

15.3 POSTULATED ACCIDENTS (CLASS 2 ACCIDENTS)

Class 2 accidents include those which

- (1) may induce fuel failures,
- (2) may lead to a breach of barriers and fission product release,
- (3) may require operation of engineered safety features, or
- (4) may result in offsite radiation exposures in excess of normal operational limits.

The events discussed in this section are assumed to occur infrequently and are not required to meet the SAFDLS. The ultimate criteria applied to these transients is a radiation exposure limit. Events in this category are:

- 15.3.1 Loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuates emergency core cooling system.
- 15.3.2 Minor secondary system pipe breaks.
- 15.3.3 Inadvertent Loading Operation of a fuel assembly into the improper position.
- 15.3.4 Seized Rotor Event.

Reanalysis for event 15.3.3 has not been required since the discussion presented in that section remains applicable.

Events 15.3.1 and 15.3.4 were reanalyzed for the extended power uprate.

15.3.1 LOSS OF REACTOR COOLANT FROM SMALL RUPTURED PIPES OR FROM CRACKS IN LARGE PIPES WHICH ACTUATES EMERGENCY CORE COOLING SYSTEM

15.3.1.1 Identification of Causes

A loss of coolant accident is defined as a rupture of the reactor coolant system piping.

Should a small break occur, depressurization of the reactor coolant system causes the pressurizer to empty. This results in a decrease of both the reactor coolant pressure and the pressurizer level. A reactor trip occurs when the low thermal margin trip setting is reached. Safety injection is then initiated by low reactor coolant pressure. The high containment pressure signal is used as a backup for a reactor trip and safety injection actuation signal. The safety injection system provides borated water to reflood and recover the reactor core to limit clad temperatures. When the reactor coolant system pressure falls below 245 psia, the safety injection tanks begin to inject water into the reactor coolant system.

A containment isolation signal (CIS) is actuated by high containment pressure or radiation. The CIS will initiate closure of all non-essential containment isolation valves and will start the shield building ventilation system.

The actuation response time of the SIAS and the CIS instrumentation is discussed in Section 7.3.2.1.

15.3.1.2 Analysis of Effects and Consequences for Small Break LOCA

The postulated SBLOCA event is defined as a break in the RCS pressure boundary which has an area of up to approximately 10% of the cold leg pipe area. The most limiting break location is in the cold leg pipe on the discharge side of the RCP. This break location results in the largest amount of inventory loss and the largest fraction of ECCS fluid being ejected out through the break. This produces the greatest degree of core uncover, the longest fuel rod heatup time, and consequently, the greatest challenge to the 10 CFR 50.46(b)(1-4) criteria.

The SBLOCA event is characterized by a slow depressurization of the primary system with a reactor trip occurring on a Low Pressurizer Pressure signal. The SIAS occurs when the system has further depressurized. The capacity and shutoff head of the HPSI pumps are important parameters in the SBLOCA analysis. For the limiting break size, the rate of inventory loss from the primary system is such that the HPSI pumps cannot preclude significant core uncover. The primary system depressurization rate is slow, extending the time required to reach the SIT pressure or to recover core liquid level on HPSI flow. This tends to maximize the heatup time of the hot rod which produces the maximum PCT and local cladding oxidation. Core recovery for the limiting break begins when the SI flow that is retained in the RCS exceeds the mass flow rate out the break followed by injection of SIT flow. For very small break sizes, the primary system pressure does not reach the SIT pressure.

The Framatome (formerly AREVA) S-RELAP5 SBLOCA evaluation model for event response of the primary and secondary systems and hot fuel rod used in this analysis (Reference 113) consists of two computer codes. The appropriate conservatisms, as prescribed by Appendix K of 10 CFR 50, are incorporated. This methodology has been reviewed and approved by the NRC to perform SBLOCA analyses.

1. The RODEX2-2A code was used to determine the burnup-dependent initial fuel rod conditions for the system calculations.
2. The S-RELAP5 code was used to predict the thermal-hydraulic response of the primary and secondary sides of the reactor system and the hot rod response.

The RODEX2-2A gap conditions used to initialize S-RELAP5 are taken at EOC, consistent with an EOC top-peaked axial power distribution. The use of EOC fuel rod conditions along with an EOC power shape is bounding of BOC because (1) the gap conductance is higher at EOC, (2) the power shape is more top-skewed at EOC, and (3) the initial stored energy, although higher at BOC, has a negligible impact on the SBLOCA results since the stored energy is dissipated long before core uncover.

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15.3.1.3 SBLOCA Results

The SBLOCA analysis was performed with the EPU core power of 3020 MWt plus 0.3% measurement uncertainty. The analysis was performed to support 10% Steam Generator tube plugging with an asymmetry of $\pm 2\%$ and an RCS flow rate of 375,000 gpm. The analysis supports a radial peaking factor (F_r) of 1.65 (1.749 with uncertainties), and a maximum Linear Heat Rate (LHR) of 15.0 kW/ft. The HPSI system was modeled to deliver the total SI flow asymmetrically to the broken loop and three intact loops as described in Table 15.3.1-1.

Table 15.3.1-2 displays the parameter values assumed in the current analysis. Analysis of the limiting SBLOCA (3.70 in. diameter) with the parameters specified in Table 15.3.1-2 produced a peak cladding temperature (PCT) of 1828^{oF(1)}. The peak local oxidation (PLO) was 3.31%, which is for the 3.6 in. and 3.7 in. diameter break. The maximum core-wide oxidation (CWO) was 0.041%, which is for the 3.6 in. diameter break.

SBLOCA break spectrum calculations were performed for break diameters of 3.5, 3.6, 3.65, 3.7, 3.75 and 3.8 in. The limiting break diameter was determined to be 3.70 in. From Figure 15.3.1-14, it can be observed that the increase in cladding temperature was being mitigated by HPSI flow just prior to the cladding being quenched by SIT flow. This typifies the limiting case. Break diameters larger than 3.8 in. are less limiting due to faster RCS depressurization and earlier timing of SIT flow. For smaller break diameters, the break flow becomes smaller such that the ability of the HPSI flow to maintain RCS mass and limit core uncover becomes greater, which will result in lower PCTs.

The plant responses to the transient for the limiting break diameter case (3.70 in) are shown in Figures 15.3.1-1 through 15.3.1-14.

Additional calculations were performed to evaluate a break in the Safety Injection Tank (SIT) line, and to evaluate the impact of delayed reactor coolant pump (RCP) trip. The SIT line break results demonstrate that it is non-limiting compared to the break spectrum results.

The calculation of record for this analysis is in Section 15.5, Reference (8).

RCA Trip Sensitivity Calculations

[A RCP trip study is performed to validate the conclusions of the earlier CEOG study (CEN-268, Revision 1 & Supplement 1-P, Revision 1-P), approved by the NRC (ML031150282), for applicability to EPU operation.]

The SBLOCA break spectrum calculations conservatively assumed RCP trip at reactor trip due to an assumed loss of offsite power at reactor trip. However, a delayed RCP trip following loss of subcooling margin (or reactor coolant system pressure of 1600 psia) can potentially challenge the 10 CFR 50.46 criteria. Continued operation of the RCPs can result in more integrated mass lost out the break and also tends to maintain RCS pressure at a plateau until the RCPs are tripped. This could potentially result in a reduced HPSI flow rate early in the transient. The combined effect could result in less RCS and RV mass, and more core uncover, challenging the PCT criterion.

Series of calculations using Appendix K models was performed for both cold leg and hot leg break cases with various RCP trip delay times. Results from the hot leg break cases were more limiting than the results from the cold leg break cases. The results of the hot leg break delayed RCP trip cases indicate that there is at least 2 minutes to trip all four RCPs after the RCS pressure reaches 1600 psia (loss of subcooling margin is lost prior to the RCS pressure reaching 1600 psia) in order to meet the 10 CFR 50.46 criteria.

A delayed RCP trip analysis was also performed using conservative best estimate models. Both cold leg and hot leg break cases with various RCP trip delay times were analyzed. The results indicate there is at least 15 minutes to trip all four RCPs after the RCS pressure reaches 1600 psia (loss of subcooling margin is lost prior to the RCS pressure reaching 1600 psia) in order to meet the 10 CFR 50.46 criteria.

It is seen that the RCP study, performed for St. Lucie Unit 1 EPU operation, supports the same conclusions as stated in the CEOG study and the corresponding NRC SE using similar approach. Thus, no changes to plant procedures, with respect to RCP trip, are needed.

(1) The PCT changes due to evaluation model errors/changes are documented in the 10 CFR 50.46 reports. These PCT changes are not (or may not be) reflected in the PCT documented here.

TABLE 15.3.1-1

HPSI FLOW RATE VS. RCS PRESSURE USED IN THE SBLOCA EVENT

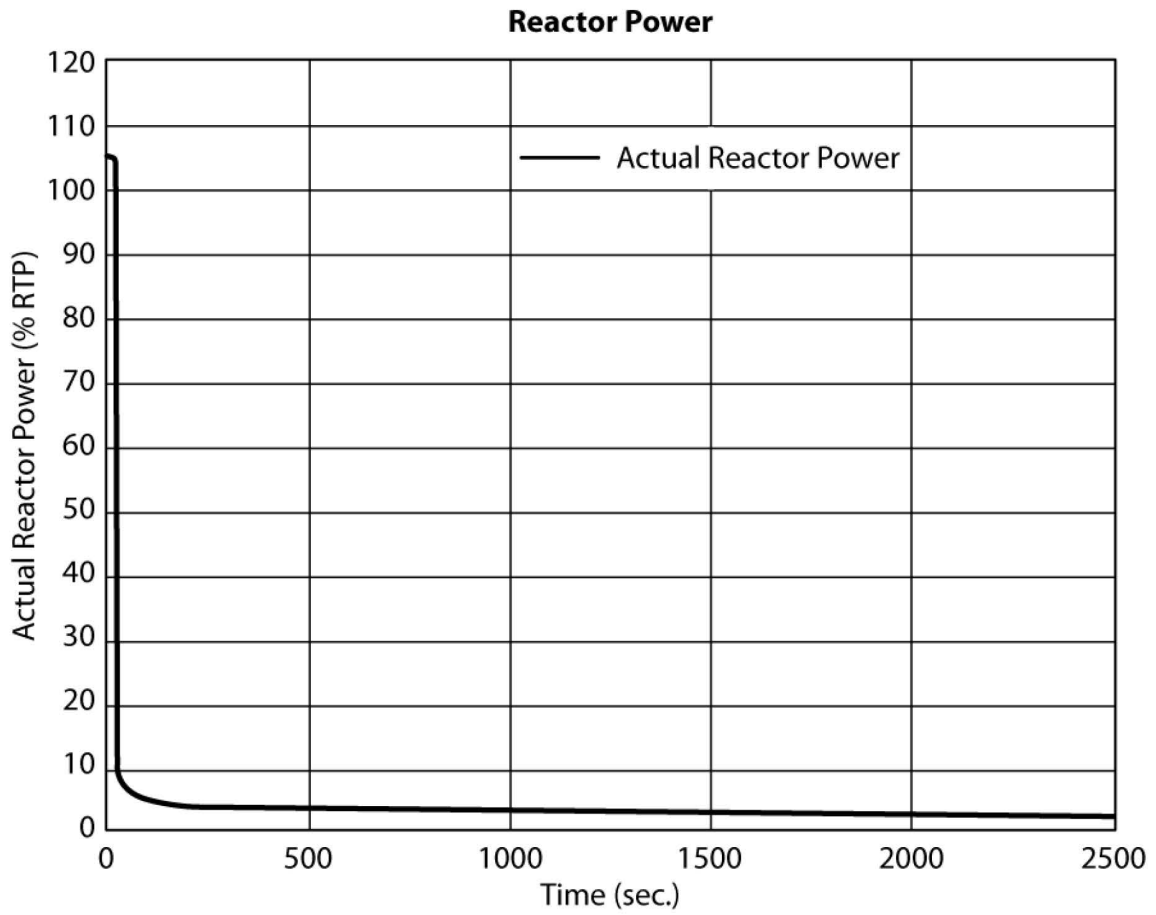
RCS Pressure (psia)	Broken Loop (gpm)	Intact Loop (gpm)
15	160.6	151.9
315	138.4	130.97
615	110.6	104.6
815	88.6	83.73
1015	57.3	54.2
1115	32.0	29.17
1125	27.3	24.9
1135	17.8	16.2
1145	1.8	1.633
1145.5	0.0	0.0

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Table 15.3.1-2

CURRENT SBLOCA ANALYSIS PARAMETERS

Parameter	Current Analysis Value
Reactor Power, MWt	3020 + 0.3% measurement uncertainty
Peak LHR, kW/ft	15.0
Radial Peaking Factor (1.65 plus uncertainty)	1.749
RCS Flow Rate, gpm	375000
Pressurizer Pressure, psia	2250
Core Inlet Coolant Temperature, °F	551
SIT Pressure, psia	244.7
SIT Fluid Temperature, °F	120
SG Tube Plugging Level, %	10
SG Secondary pressure, psia	830
MFW Temperature, °F	436
AFW Temperature, °F	111.5
Low SG Level AFAS Setpoint, %	5
HPSI Fluid Temperature, °F	104
Charging system delay time, sec	150
Reactor Scram Low Pressurizer Pressure Setpoint, psia	1807
Reactor Scram Delay Time on Low Pressurizer Pressure, sec	0.9
Scram CEA Holding Coil Release Delay Time, sec	0.5
SIAS Activation Setpoint Pressure for harsh conditions, psia	1520
HPSI Pump Delay Time on SIAS, sec	30
MSSV lift pressures	Nominal + 3% uncertainty

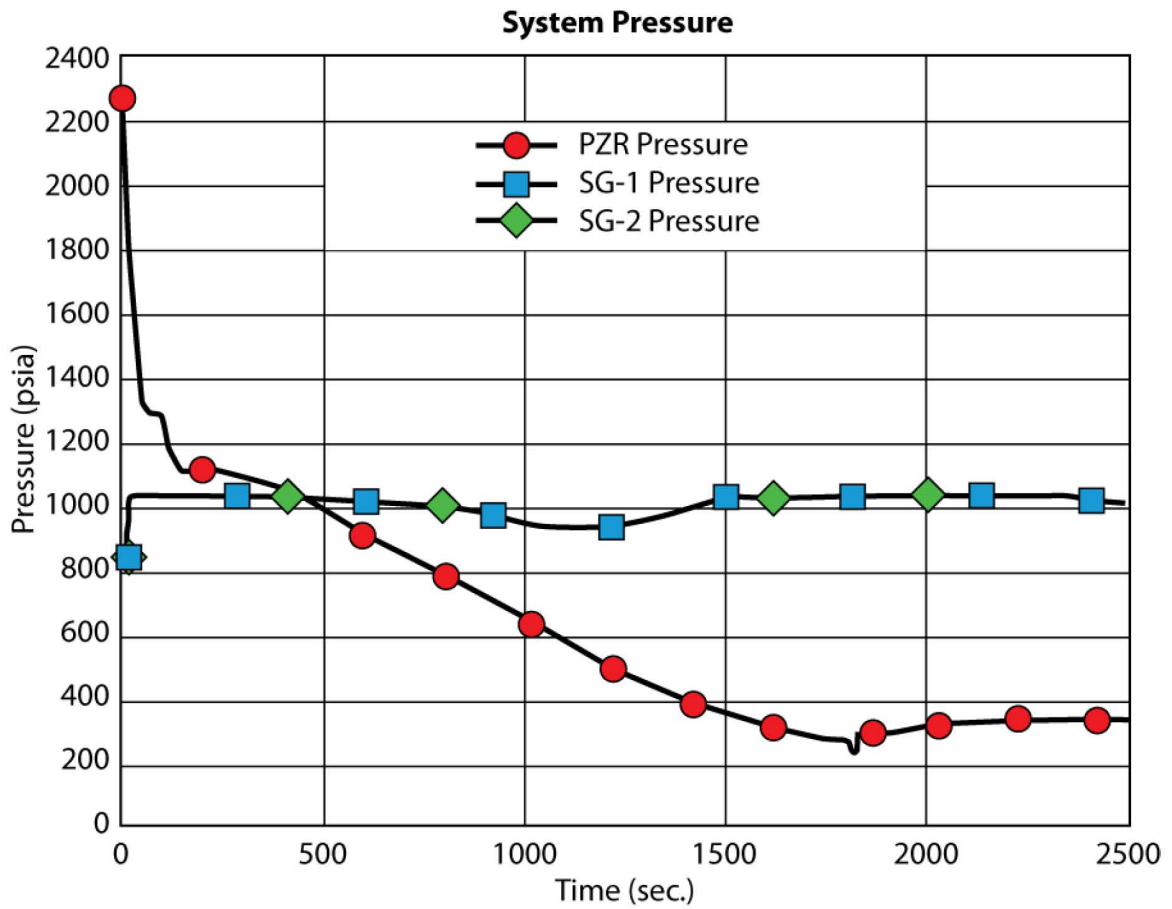


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**REACTOR POWER FOR 3.70 IN.
DIAMETER BREAK
SBLOCA**

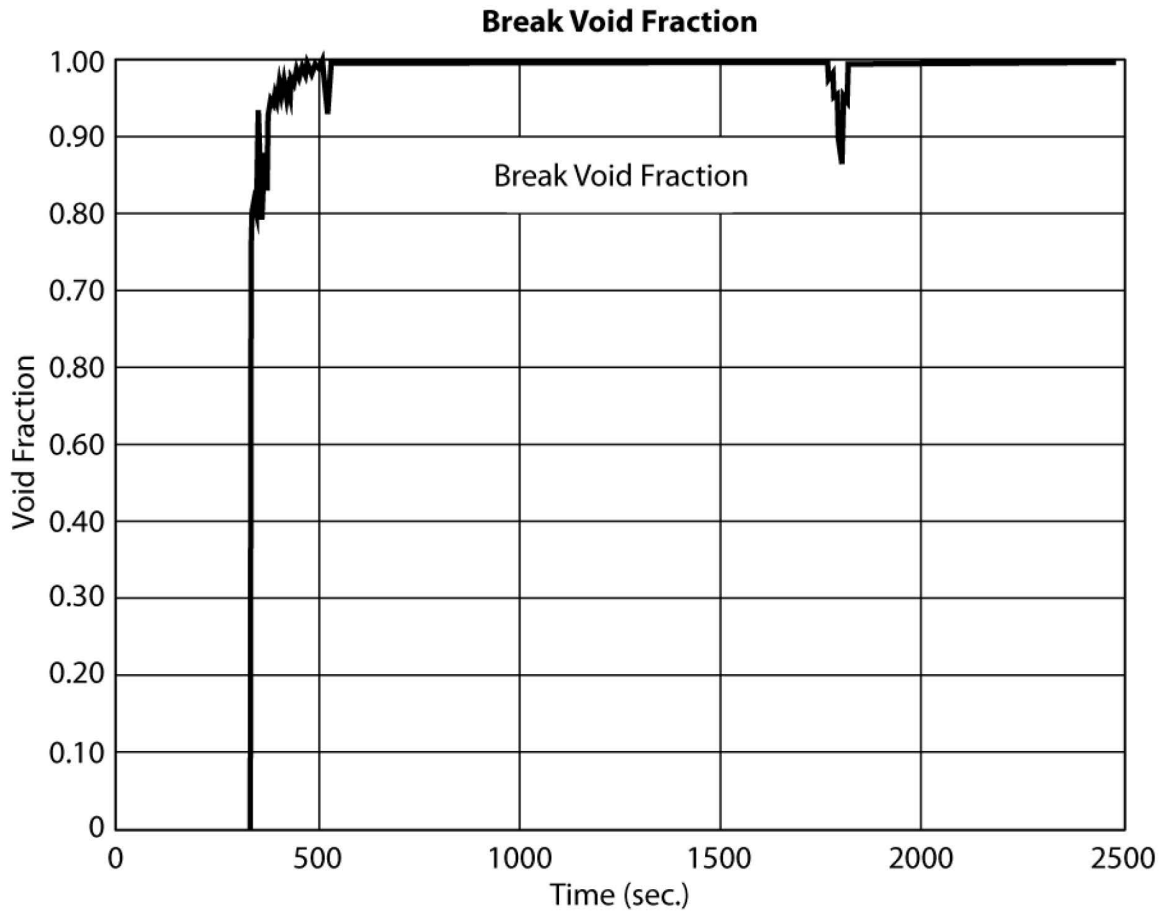
FIGURE 15.3.1-1



Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1
PRIMARY AND SECONDARY PRESSURES
FOR 3.70 IN. DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-2

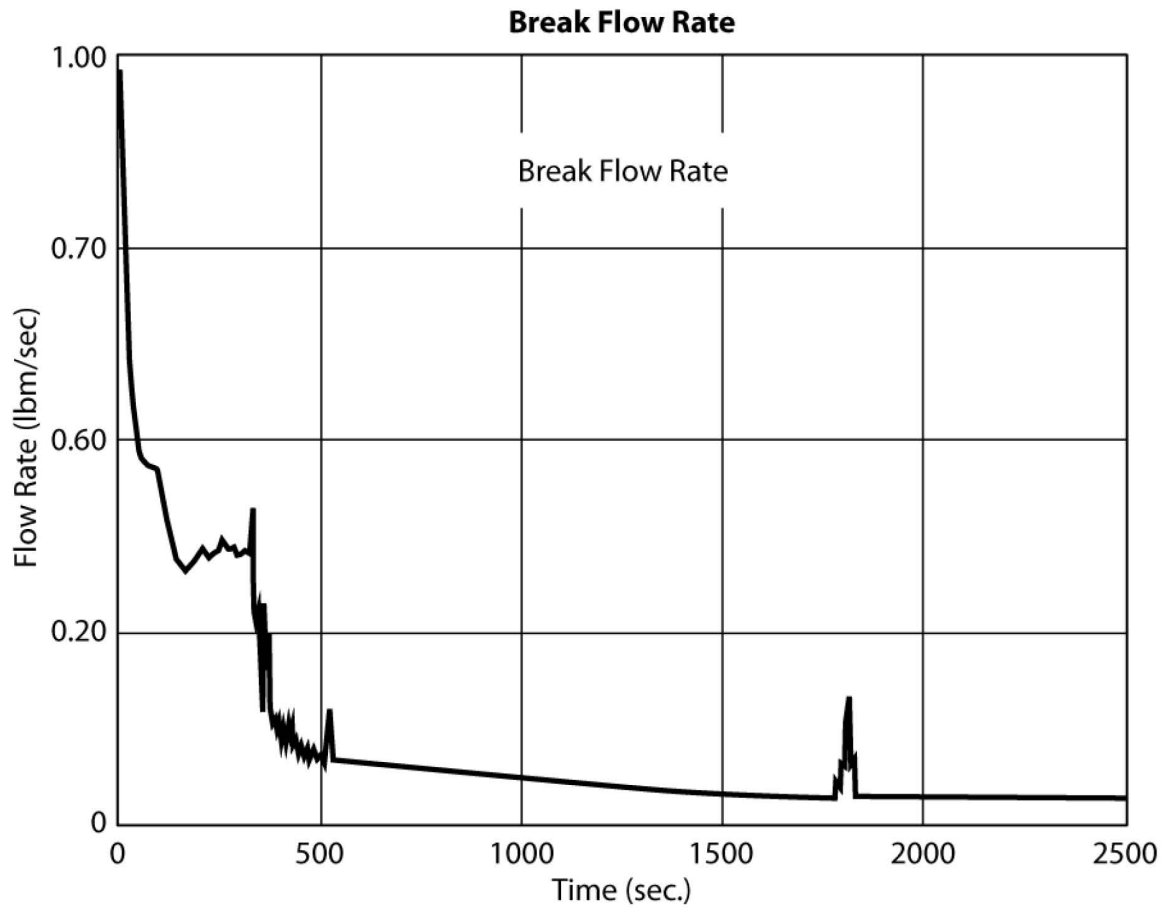


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**BREAK VOID FRACTION FOR 3.70 IN.
DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-3

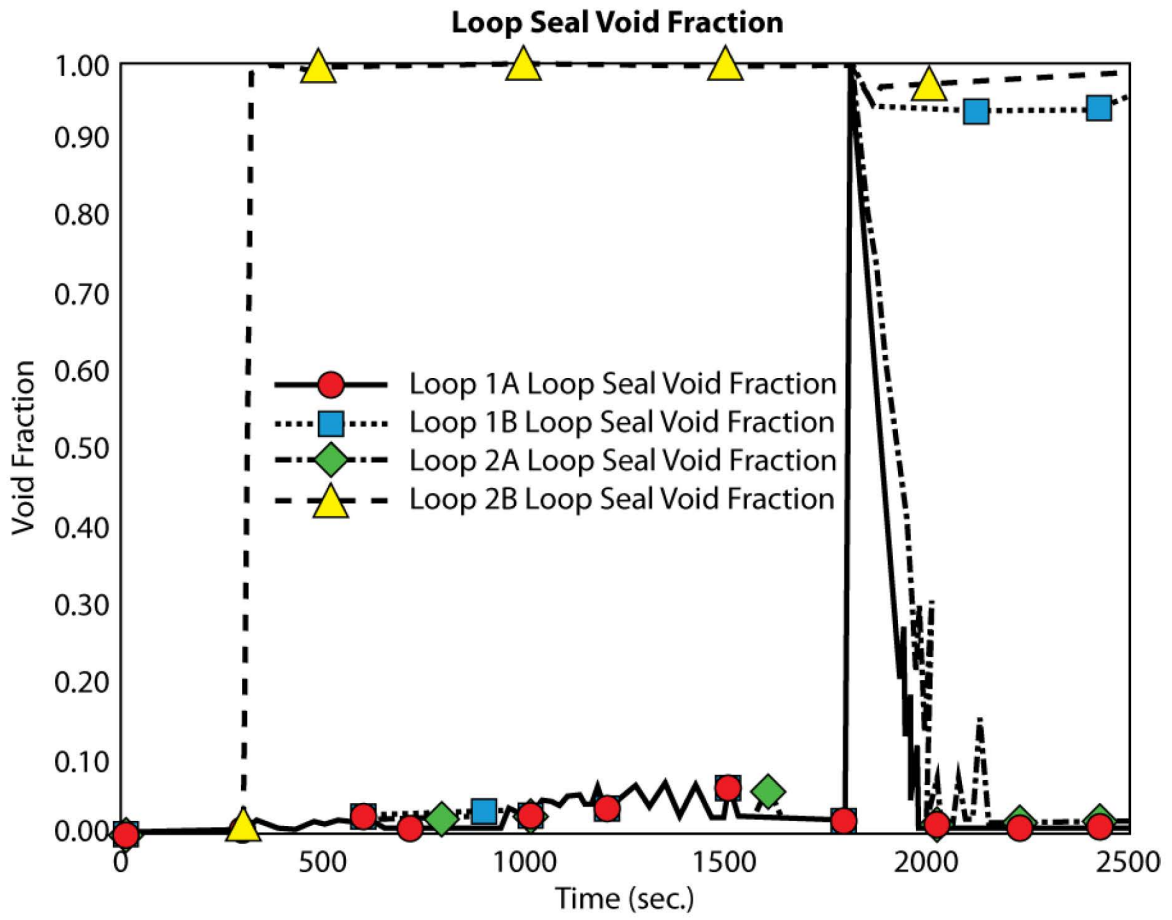


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**BREAK FLOW RATE FOR 3.70 IN.
DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-4



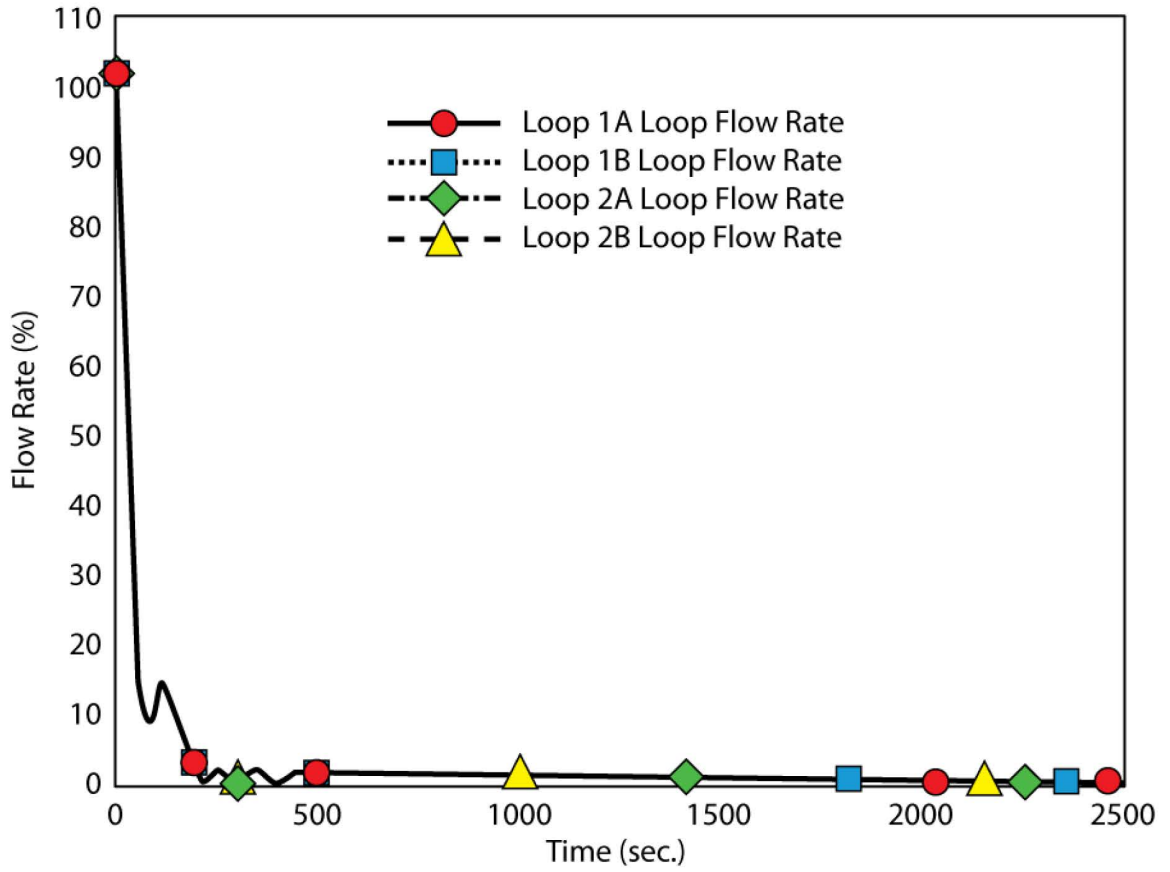
Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**LOOP SEAL VOID FRACTIONS FOR
3.70 IN. DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-5

RCS Loop Flow Rate

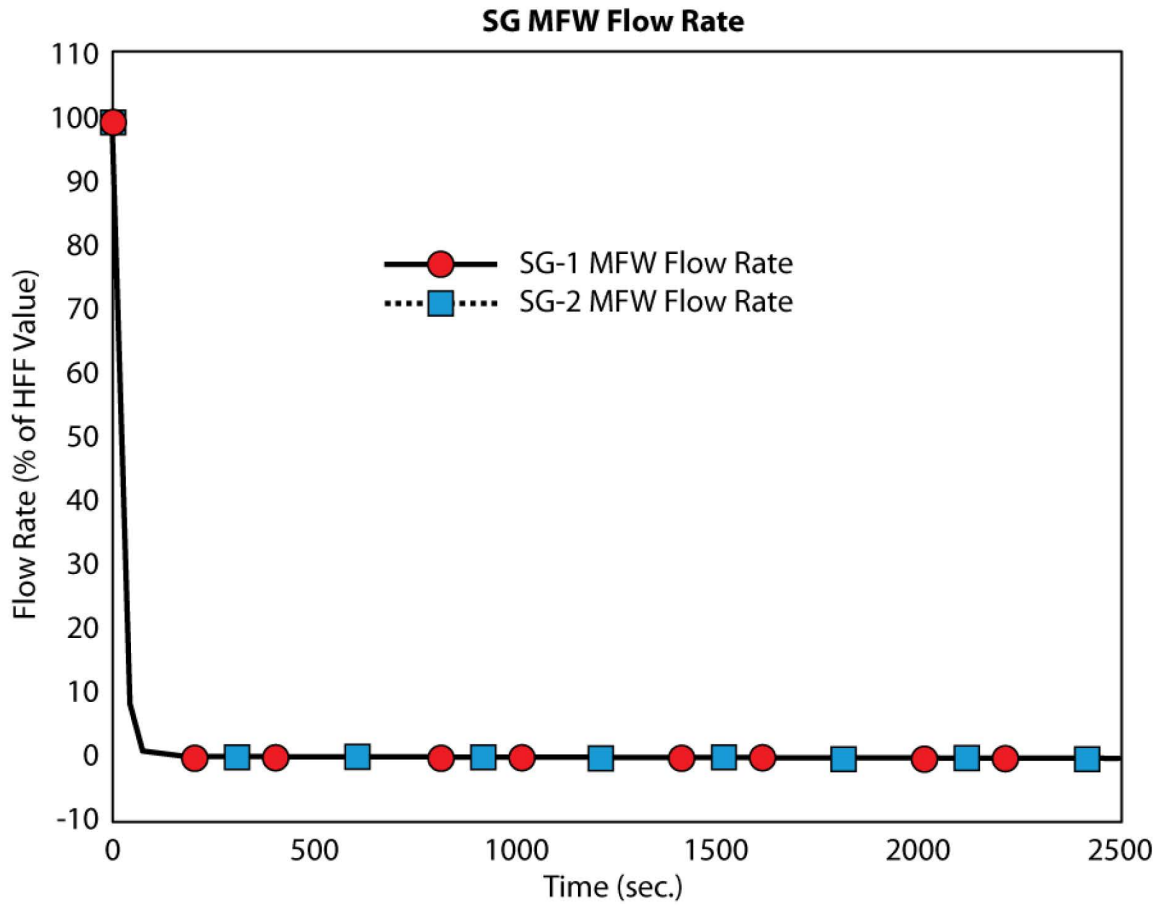


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**RCS LOOP FLOW RATE FOR 3.70 IN.
DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-6

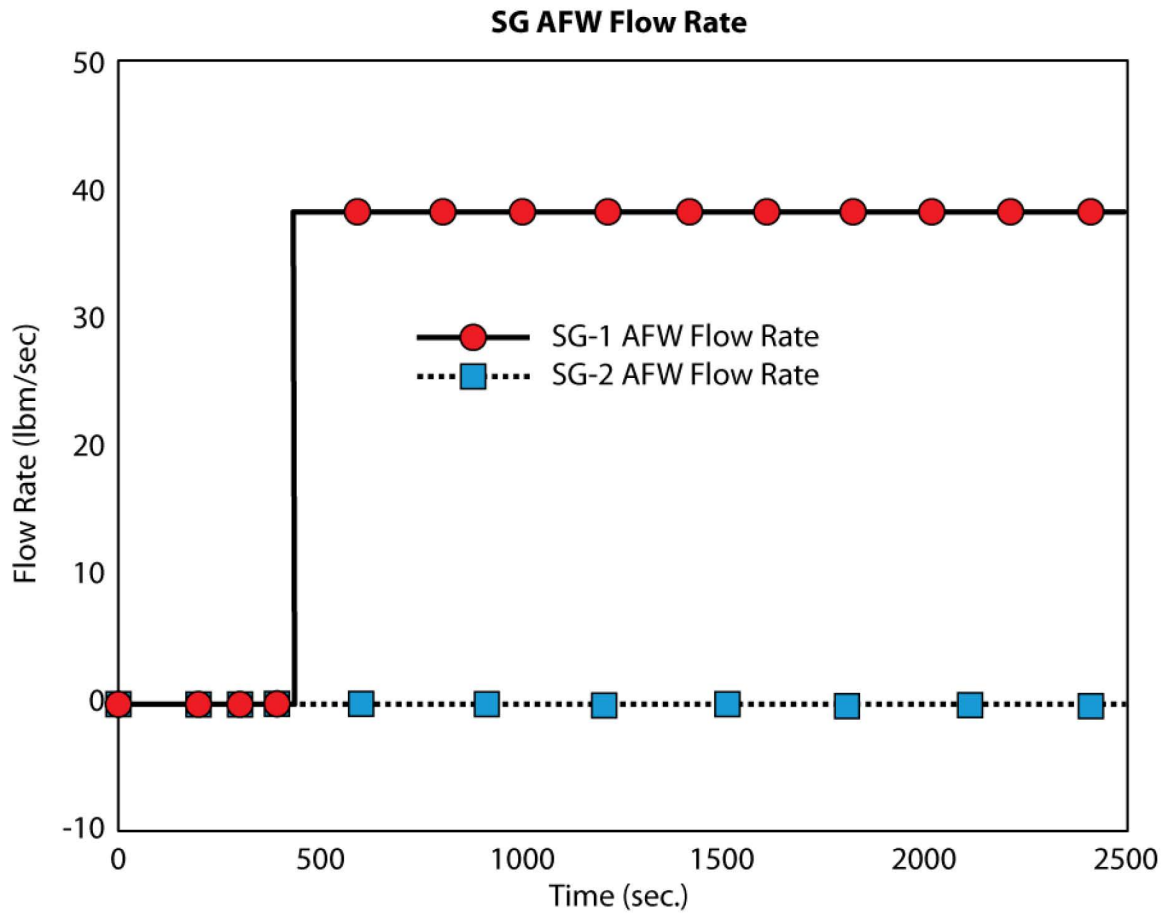


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**MFW FLOW RATE FOR 3.70 IN.
DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-7

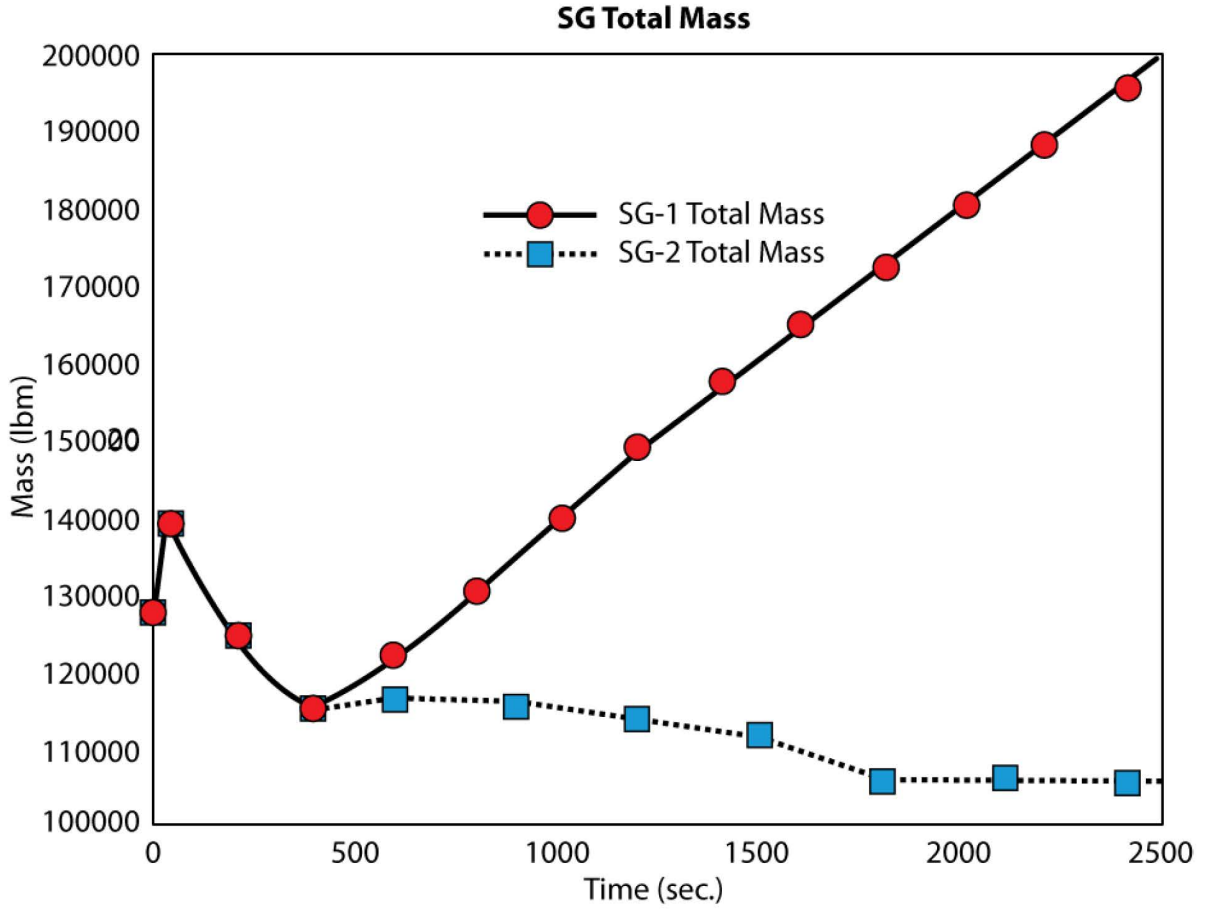


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**AFW FLOW RATE FOR 3.70 IN.
DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-8

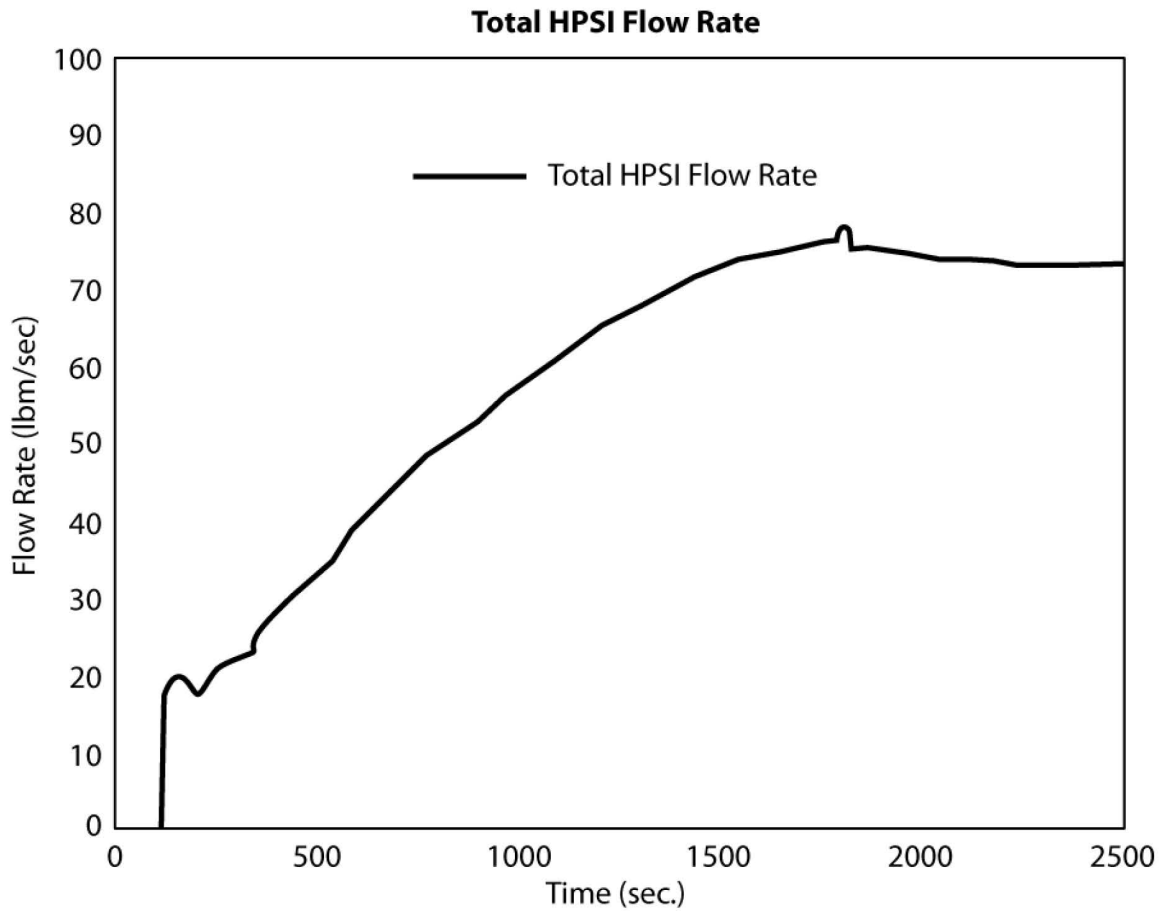


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**STEAM GENERATOR TOTAL MASS FOR
3.70 IN. DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-9

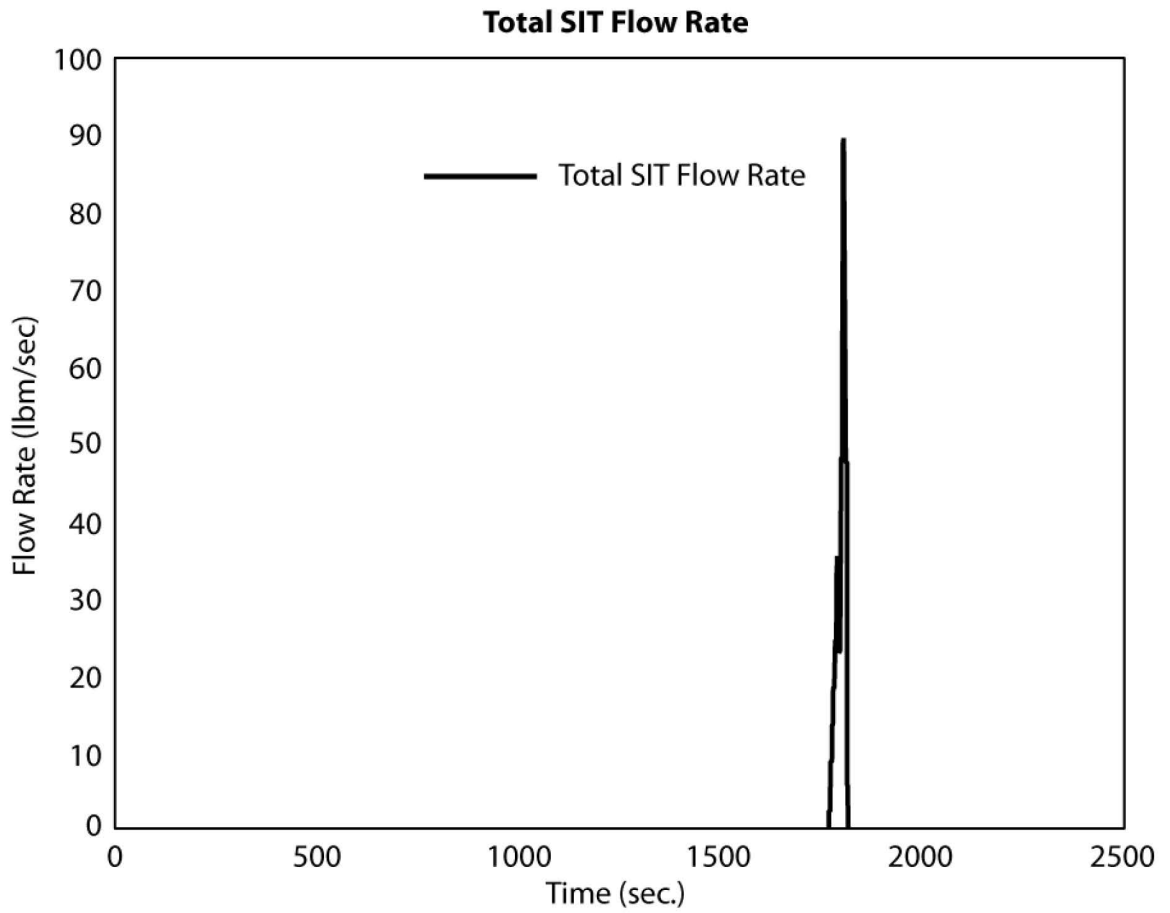


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**TOTAL HPSI MASS FLOW RATE FOR
3.70 IN. DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-10

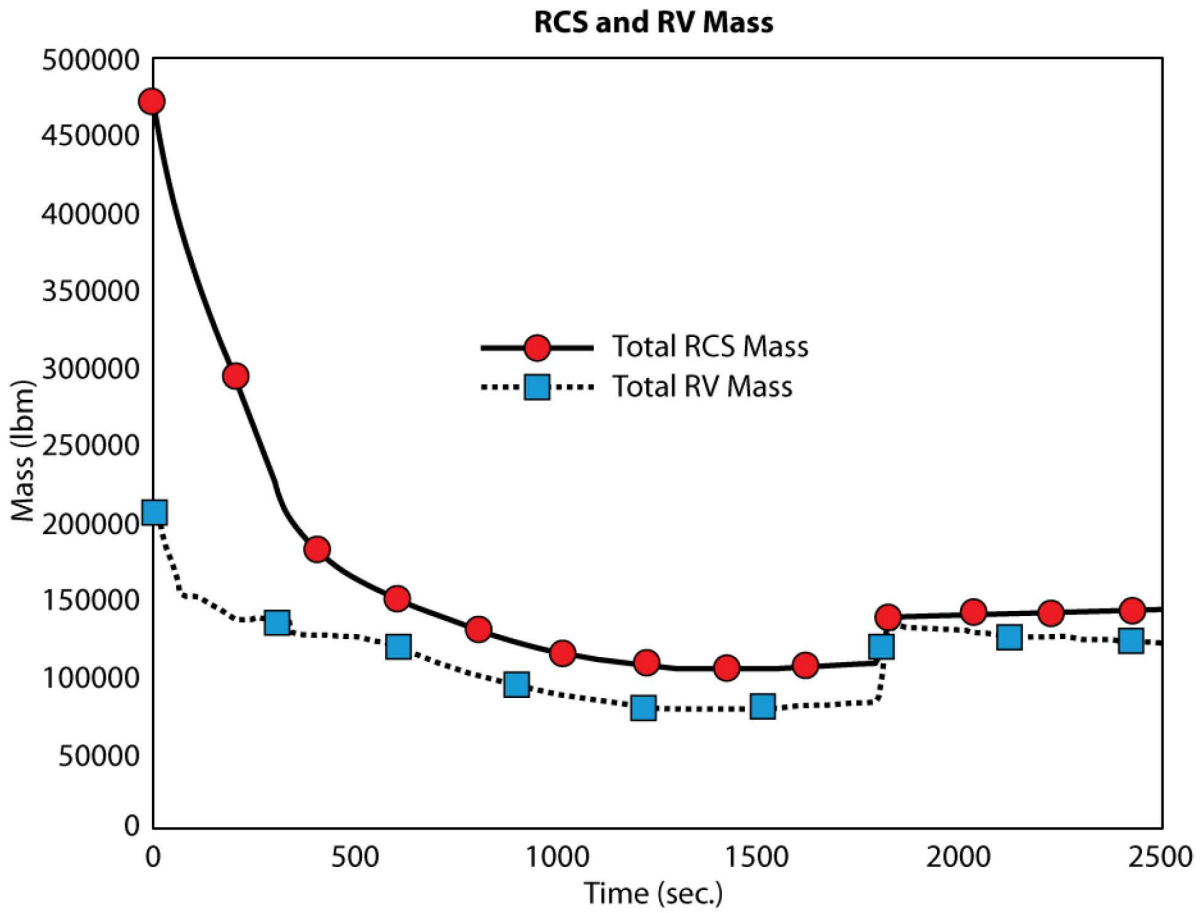


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**TOTAL SIT MASS FLOW RATE FOR
3.70 IN. DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-11

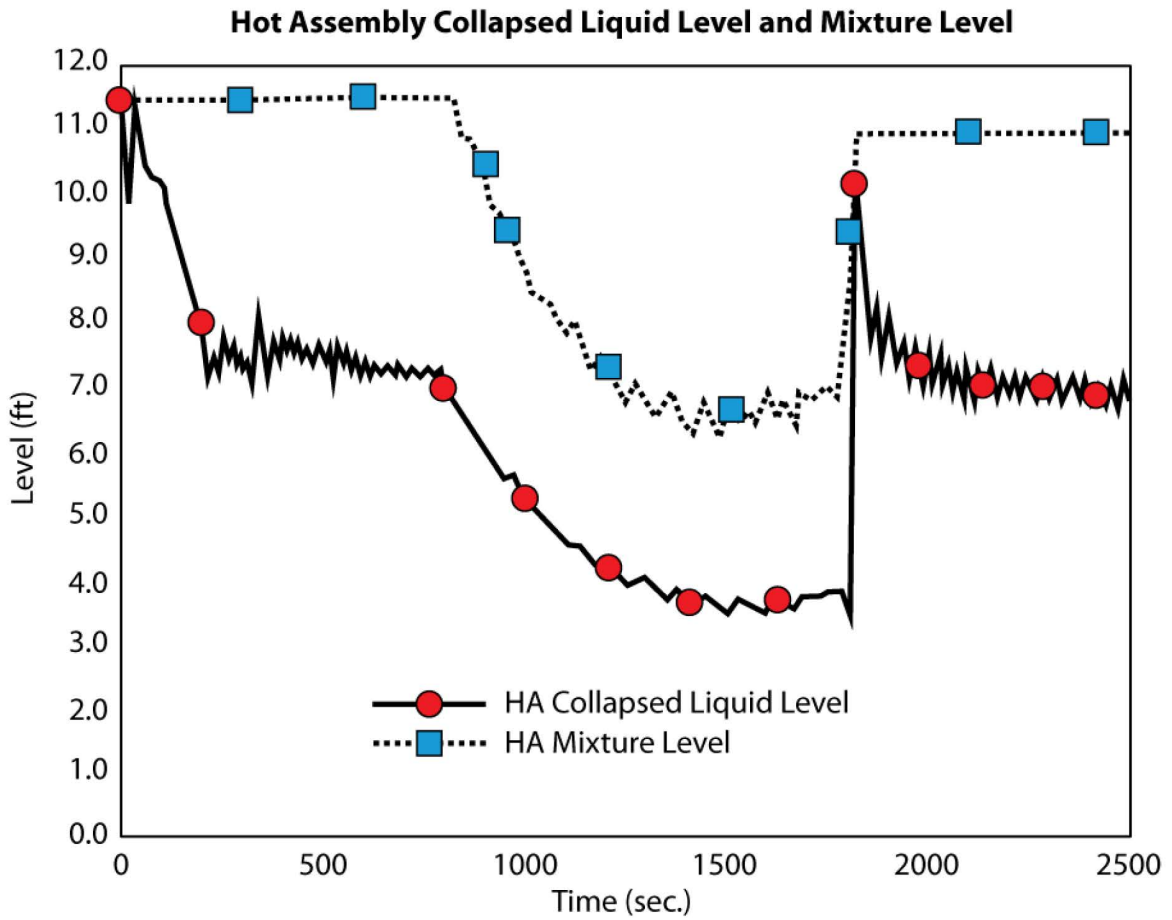


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**RCS AND REACTOR VESSEL MASS
INVENTORIES FOR 3.70 IN.
DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-12

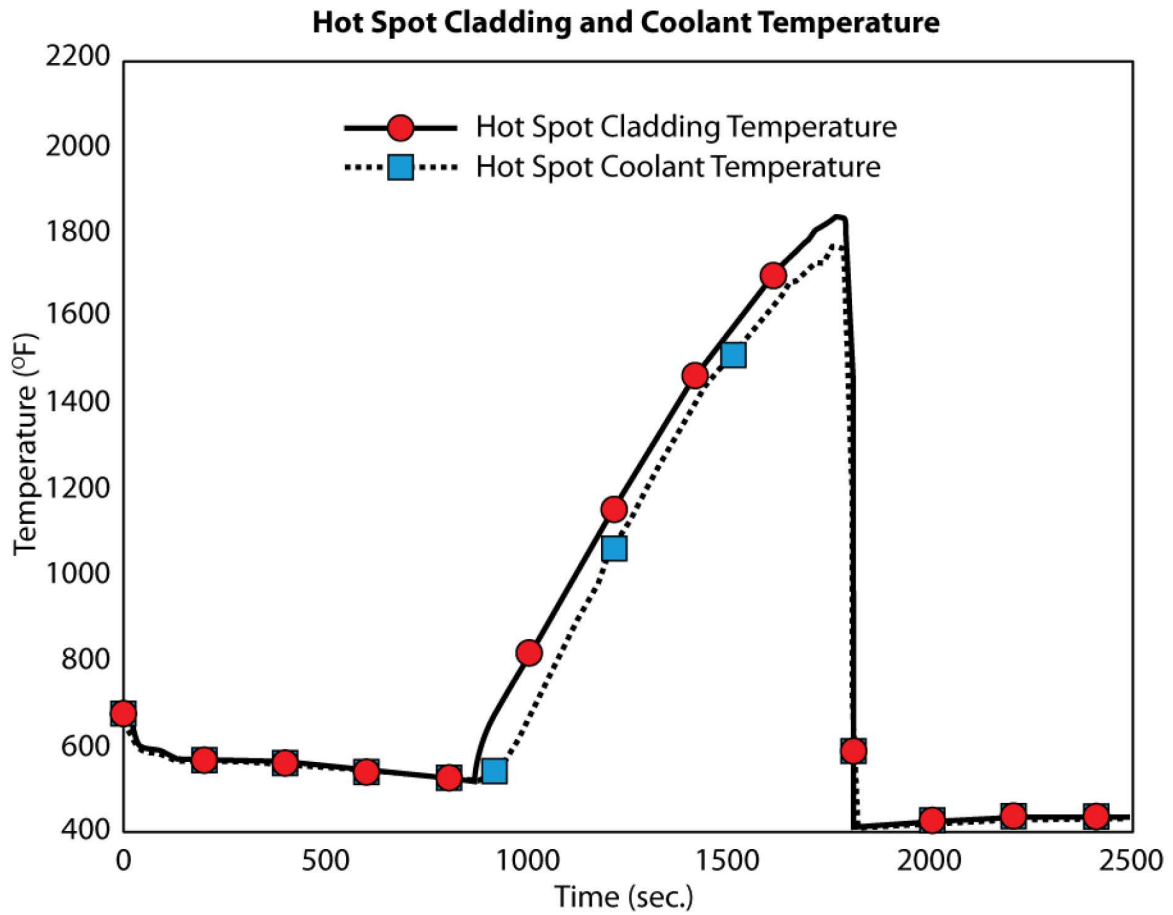


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**HOT ASSEMBLY COLLAPSED LIQUID
LEVEL FOR 3.70 IN.
DIAMETER BREAK
SBLOCA**

FIGURE 15.3.1-13



Amendment No. 27 (04/15)

FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
HOT SPOT CLADDING TEMPERATURE AND COOLANT TEMPERATURE FOR 3.70 IN. DIAMETER BREAK SBLOCA
FIGURE 15.3.1-14

15.3.2 MINOR SECONDARY SYSTEM PIPE BREAKS

15.3.2.1 Identification of Causes

A minor secondary system pipe break is defined as one which results in steam blowdown rates equivalent to a 6 inch diameter break outside the containment.

15.3.2.2 Analysis of Effects and Consequences

A separate analysis is not required for minor secondary system pipe breaks. The break area is less than the full steam line area for the main steam line break event presented in Section 15.4.6, and any fuel failure predicted for the main steam line break event will bound this event. Therefore, this event is bounded by the main stem line break event.

15.3.3 INADVERTENT LOADING OF A FUEL ASSEMBLY INTO THE IMPROPER POSITION

15.3.3.1 Identification of Causes

There are two accidents considered in this section; first, the erroneous loading of fuel pellets or fuel rods of different enrichment in a fuel assembly, and second, erroneous placement or orientation of fuel assemblies.

An error in assembly, fabrication or core loading is considered to be extremely unlikely. The extensive quality control and quality surveillance programs employed during the fabrication process, along with the strict procedural control used during core loading, will preclude the possibility of such errors. However, even if fuel rods or assemblies were incorrectly placed, these would either be detectable from the results of the startup testing program or would lead to a minimal number of rods at excessive power during operation.

15.3.3.1.1 Erroneous Loading of Fuel Pellets or Fuel Rods of Different Enrichment in a Fuel Assembly

The probability of manufacturing a fuel assembly with an incorrect enrichment is remote. The extensive quality control and quality surveillance programs in effect during the manufacture of the fuel pellets, in the pellet loading of the fuel rods, and in the fuel rod installation in the fuel assembly precludes the possibility of manufacturing a fuel assembly with incorrect enrichments.

During the manufacture and assembly of the fuel rods, numerous check points and assay tests ensure that the enrichment is as specified and that the fuel rods and fuel assemblies are properly loaded and assembled. An assay is made of each lot of UO₂ powder to ensure that the enrichment is as required by the fuel specification. During the manufacture of the pellets each powder lot is isolated during processing. In addition to batch identification, each pellet is identified with an imprinted number or letter identifying the enrichment. After sintering, an additional enrichment check of the fabricated pellets is made by random sampling. Assembly of the fuel rods is performed by loading the fuel pellets into cladding onto which one end cap has been welded. The end cap is marked prior to welding to identify the enrichment to be loaded into the fuel rod.

During fuel rod assembly the quality control procedures require verification of each fuel rod in the assembly. Each fuel assembly is identified by a serial number which is engraved on the upper end fitting as shown on Figure 4.2-4.

A record of all operations performed on each fuel rod and each poison rod up to and including final fuel assembly is recorded on computer punch cards, one for each fuel rod and each poison rod. This procedure permits a rapid check at completion of fabrication to ensure that each rod within the fuel assembly has completed all of the required steps within the fabrication process. The computerized record keeping system also has the advantage of

providing a mechanism by which an accurate record of all fuel rods within a fuel assembly can be defined as well as the enrichment, weight of uranium U-235 and UO₂, and lot number of the fuel within each fuel rod.

15.3.3.1.2 Erroneous Placement or Orientation of Fuel Assemblies

The fuel enrichment within a fuel assembly is identified by a coded serial number marked on the exposed surface of the upper end fitting of the fuel assembly. This serial number is used as a means of positive identification for each assembly in the plant. A status board is provided in the refueling control center that shows a schematic representation of the reactor core, spent fuel pool and new fuel storage area. During the period of core loading, the location of each CEA, fuel assembly and source is shown by its identification number on this status board.

The status board in the refueling control center will be constantly updated by a designated member of the reactor operations staff whenever a fuel assembly is being moved. The staff member in the refueling control center will be in constant communication with each area where this is occurring. Also, a senior licensed operator will be present during alteration of the core to directly supervise the activity and ensure that the assemblies are moved to the correct locations. Fuel assemblies will not be moved unless these lines of communication are available. In addition to these precautions, periodic independent inventories of components in the reactor core, spent fuel, and new fuel storage areas will be made to ensure that the status board is correct. Also, the exposed surfaces of the upper end fittings are inspected, either at the end of core loading or during each fuel assemblies' final move into the core, to verify that all assemblies are correctly located. These precautions are included in the core loading procedures which are to be reviewed by appropriate plant personnel.

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15.3.3.2 Analysis of Effects and Consequences

15.3.3.2.1 Erroneous Loading of Fuel Pellets or Fuel Rods of Different Enrichment in a Fuel Assembly

When the power distribution is peaked in a peripheral high enrichment C- assembly it is expected that replacing single batch C-fuel rod (2.82 w/o U235) by lower enriched fuel from batches A (1.93 w/o U235) or B (2.33 w/o U235) would not cause any peaking problems. In order to increase peaking in the periphery, fuel rods of a higher enrichment than the highest (2.82 w/o) used in batch C would have to be inserted into a C-assembly. This is not considered to be credible.

Another possible combination would be the replacement of A-fuel rods by C-fuel rods. The magnitude of the local peaking factor increase would depend on several conditions that must be arbitrarily postulated, such as the number of A-fuel rods replaced by C-fuel rods and the location of these rods with respect to the water holes. At one end of the scale of postulated situations, the increased local peaking would be too small to adversely affect core performance. These situations may go undetected. Fuel loading errors that cause local power peaking increases large enough to adversely affect core performance will be detectable through the use of incore instrumentation during startup testing.

15.3.3.2.2 Erroneous Placement or Orientation of Fuel Assemblies

If, in spite of the above precautions to prevent misloading a fuel assembly into the reactor core, it is assumed that an assembly is placed in the wrong core position, then many combinations may exist. Probably the worst combination would be the interchange of two assemblies of different reactivities. This would tend to maximize the induced power tilt. Figures 15.3.3-2, 15.3.3-3 and 15.3.3-4 for a similar rated plant (Maine Yankee Docket 50-309) show examples of postulated interchanges of fuel assemblies in the core. Relative power densities are displayed for each fuel assembly in a quarter core together with maximum fuel rod peaks. The data should be compared with the unperturbed power distribution of BOL shown in Figure 15.3.3-1. Figures 15.3.3-2 and 15.3.3-3 indicate interchanges of unshimmed and shimmed batch C assemblies. Figure 15.3.3-4 shows the effect of an interchange of an unshimmed batch C assembly with the central batch A fuel assembly. The examples chosen yield appreciable radial peaking. These analyses were performed with standard design calculational tools similar to those described in Section 4.3.

As has been shown, the incorrect location of fuel assemblies can lead to an appreciable departure from the expected BOL power distribution. However, if this occurs, there will be an attendant change in the core characteristics. More specifically, the resultant core will have an asymmetric power distribution which would be detected by the incore (Section 4.3.5) and/or out-of-core (Section 7.2.1) instrumentation during startup testing. (This would be true even if the central assembly were interchanged with another.) Another means of detecting tilts induced by misplacement of fuel assemblies may be provided by the measurement of individual reactivity worths for symmetrically located CEAs during startup testing.

Assembly misorientation would not be a problem on the first core since each assembly is of itself symmetric.

15.3.3.3 Extended Power Uprate Evaluation

The UFSAR considers two sub-events:

- Erroneous Loading of Fuel Pellets or Fuel Rods of Different Enrichment in a Fuel Assembly
- Erroneous Placement or Orientation of Fuel Assemblies

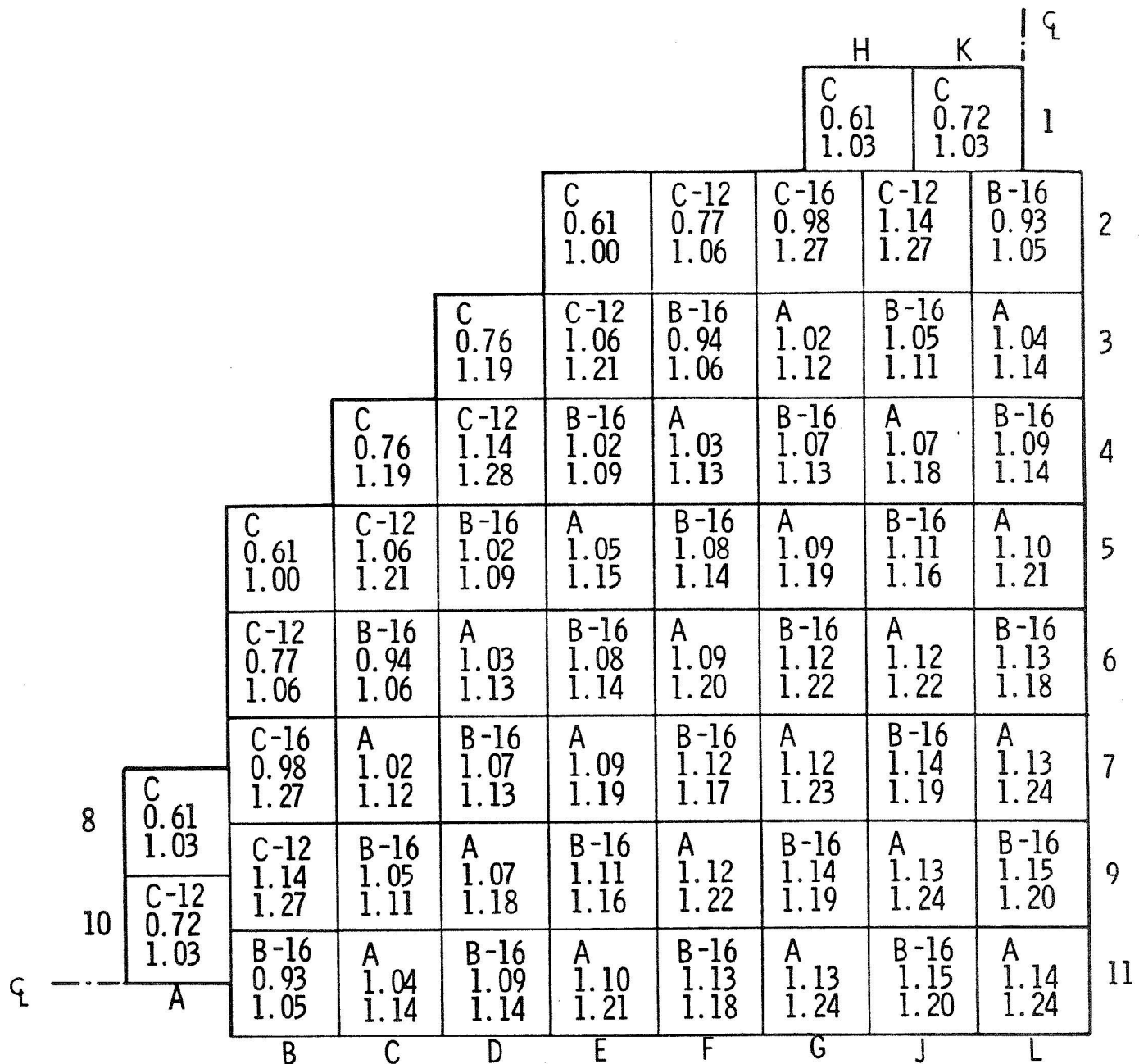
Aside from the administrative and surveillance programs specifically designed to prevent a fuel assembly mislead, several redundant systems are present to detect power maldistributions. It has been determined that with the incore detector monitoring system operable per UFSAR Section 4.2.2.2.8 and the Technical Requirements Manual, the detection of misloaded assemblies is not adversely impacted by EPU.

With 75% of the detectors operational during startup, the core remains adequately monitored.

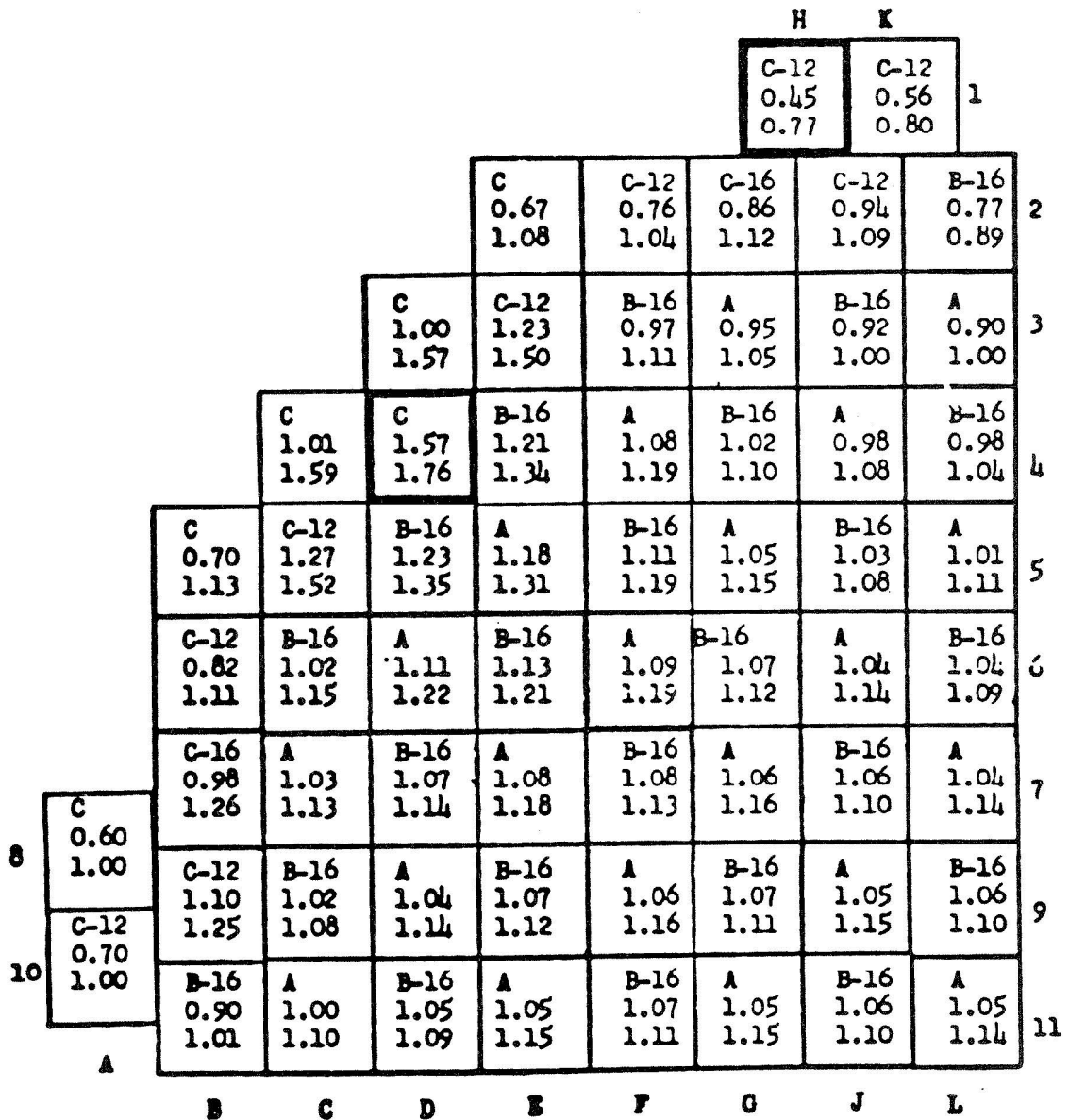
Neither of the two identified events are affected by the EPU; therefore, the current licensing analysis given in the UFSAR Sections 15.3.3.2.1 and 15.3.3.2.2 bounds the EPU.

Additionally, neither of these events are affected by placing certain irradiated fuel assemblies into dry storage casks. The Updated FSAR analyses bound Unit 1 operation with irradiated fuel in dry storage.

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478



B-16	Type of Fuel
1.08	Average Power of Assembly
1.14	Max. Rod Peak in Assembly



A	Type of Fuel
1.06	Average Power of Assembly
1.16	Max. Rod Peak in Assembly

						H	K		
						C	C-12		1
						1.14	1.40		
						1.99	2.02		
				C	C-12	C-16	C	B-16	2
				0.64	0.97	1.56	2.14	1.73	
				1.07	1.42	2.08	2.40	1.88	
			C-12	C-12	B-16	A	B-16	A	3
			0.61	1.02	1.09	1.40	1.62	1.64	
			0.97	1.22	1.30	1.66	1.80	1.82	
		C	C-12	B-16	A	B-16	A	B-16	4
		0.61	0.94	0.94	1.09	1.26	1.36	1.42	
		0.96	1.09	1.06	1.22	1.45	1.53	1.58	
	C	C-12	B-16	A	B-16	A	B-16	A	5
	0.47	0.84	0.84	0.94	1.05	1.14	1.23	1.24	
	0.78	0.98	0.93	1.04	1.16	1.26	1.35	1.37	
	C-12	B-16	A	B-16	A	B-16	A	B-16	6
	0.59	0.74	0.84	0.93	0.99	1.08	1.11	1.14	
	0.81	0.84	0.93	1.01	1.10	1.17	1.23	1.23	
	C-16	A	B-16	A	B-16	A	B-16	A	7
	0.74	0.78	0.85	0.91	0.98	1.02	1.07	1.07	
	0.96	0.87	0.92	1.00	1.05	1.12	1.14	1.17	
8	C	B-16	A	B-16	A	B-16	A	B-16	9
	0.45	0.80	0.85	0.91	0.95	1.00	1.02	1.04	
	0.77	0.86	0.93	0.97	1.04	1.06	1.12	1.09	
10	C-12	B-16	A	B-16	A	B-16	A	B-16	11
	0.53	0.79	0.85	0.90	0.95	0.98	1.02	1.02	
	0.77	0.87	0.91	0.98	1.01	1.08	1.07	1.11	
	A	B	C	D	E	F	G	J	L

C-12	Type of Fuel
0.53	Average Power of Assembly
0.77	Max. Rod Peak in Assembly

15.3.4 SEIZED ROTOR EVENT

15.3.4.1 Identification of Causes

The seized rotor event is analyzed to demonstrate that the RCS pressure limit of 2750 psia will not be exceeded and only a small fraction of fuel pins are predicted to fail during this event.

The single reactor coolant pump shaft seizure is postulated to occur as a consequence of a mechanical failure. The single reactor coolant pump shaft seizure results in a rapid reduction in the reactor coolant flow to the three-pump value. A reactor trip for the seized rotor event is initiated by a low coolant flow rate as determined by a reduction in the sum of the steam generator hot or cold leg pressure drops. This signal is compared with a setpoint.

15.3.4.2 Analysis of Effects and Consequences

The Reactor Coolant Pump Rotor Seizure (also known as Locked Rotor) event is postulated as the instantaneous seizure of a (single) reactor coolant pump rotor. Flow through the faulted RCS loop rapidly decreases, causing a reactor trip on a Low RCS Flow signal within 1 to 2 seconds and a turbine trip on the reactor trip. Furthermore, a loss of offsite power is assumed to occur at that time, which causes the remaining reactor coolant pumps to begin coasting down.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the reactor coolant. The combination of the relatively high fuel rod surface heat fluxes, decreasing core flow, and increasing core coolant temperatures challenges the DNBR safety limit and may result in fuel failure.

At the same time, the steam generator primary-to-secondary heat transfer rate decreases because (1) the decreasing primary coolant flow degrades the steam generator tube primary side heat transfer coefficients and (2) the turbine trip causes the secondary-side temperature to increase. Decreasing rate of heat removal in the steam generators combined with decreasing flow of coolant removing heat from the reactor core cause the reactor coolant to heat up. The resultant reactor coolant expansion causes fluid to surge into the pressurizer and an increase in RCS pressure. This will actuate the automatic pressurizer spray system and may even open the pressurizer PORVs and safety valves. The event may challenge the RCS overpressure criterion.

Since the systems designed to mitigate this event (namely, the RPS) are redundant, there is no single active failure that will adversely affect the consequences of the event. There are also no relevant operator actions within the (short term) time period of this analysis.

Detailed analyses were performed with approved methodologies using the S-RELAP5 and XCOBRA-IIIC codes (References 117, 108, and 109). The S-RELAP5 code was used to model the key system components and calculate neutron power, fuel thermal response, surface heat transport, and fluid conditions (such as coolant flow rates, temperatures, and pressures) and produce an estimated time of MDNBR. The core fluid boundary conditions and average rod surface heat flux were then input to the XCOBRA-IIIC code, which was used to calculate the MDNBR using the HTP CHF correlation (Reference 110).

A single case was analyzed at BOC HFP initial conditions, maximum Technical Specifications core inlet temperature and minimum Technical Specifications RCS flow rate. The factors affecting scram time were chosen to maximize the scram delay with the trip signal delay and holding coil delay times set to their maximum values. This produced the most significant challenge to the DNB limit. Note that only the DNB case is presented because the maximum RCS pressure case is not limiting. Steam release data were calculated to cool the plant down to 212°F based on an assumed operator action time of up to 45 min.

The input parameters and biasing for the analysis of this event are shown in Table 15.3.4-1. The input parameters and biasing were consistent with the approved methodology.

- Initial Conditions - HFP initial conditions, maximum Technical Specifications core inlet temperature and minimum Technical Specifications RCS flow rate were assumed in order to minimize the initial margin to DNB.
- Core Inlet Flow Distribution - A conservatively biased inlet flow rate to the limiting fuel assembly and a highly limiting gradient flow distribution were used in the DNB calculation to account for cross flow into the affected quadrant in the lower plenum caused by the seized RCP rotor.
- Reactivity Feedback - Since the event involves an increase in the core coolant temperature, the event was assumed to occur at BOC with a maximum TS MTC at full power. However, this event occurs quickly and is generally not sensitive to neutronics parameters. A minimum HFP scram worth was used to conservatively prolong the degradation in flow while maintaining relatively high core power.
- Reactor Protection System Trips and Delays - The event is primarily protected by the low flow RPS trip. The reactor protection system trip setpoints and response times were conservatively biased to delay the actuation of the trip function. In addition, a maximum CEA holding coil delay was assumed.
- Loss of Offsite Power - Loss of offsite power at the time of turbine trip (resulting from reactor trip) was assumed. The remaining three RCPs were assumed to begin to coast down with the loss of offsite power which conservatively reduced the flow for the calculation of DNB. The coastdown characteristics of the RCPs were conservatively bench marked to plant data.
- Gap Conductance - Gap conductance was set to a conservative BOC value to delay the transfer of heat from the fuel rod to the coolant allowing the primary system flow to decay further thus leading to a conservative prediction of DNBR.

This event is classified as a Postulated Accident. The principally challenged acceptance criterion for this event is with respect to radiological consequences. The extent of fuel failure was determined by the S-RELAP5 and XCOBRA-IIIC analyses. In addition, steam releases were calculated based on a plant cooldown to 212°F.

This event does not represent a significant challenge to fuel centerline melting because there is no large power increase and no significant adverse power redistribution within the core. Also, the increase in RCS pressure for this event is much less severe than the result of the Loss of External Electrical Load (see Section 15.2.7).

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15.3.4.3 Results

The sequence of events is given in Table 15.3.4-2. Plots of key system parameters are shown in Figure 15.3.4-1 to Figure 15.3.4-6. The MDNBR was statistically calculated and compared to the 95/95 limit for the HTP DNB correlation. The fuel failure due to DNB remains below the limit used in the radiological consequences analysis.

The calculations of record for these analyses are listed in Section 15.5, Reference (8).

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15.3.4.3.3 Radiological Analysis

15.3.4.3.3.1 Background

This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Fuel damage may be predicted to occur as a result of this accident. Due to the pressure differential between the primary and secondary systems and assumed steam generator tube leakage, fission products are discharged from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere from the secondary coolant system through the steam generator via the ADVs and MSSVs. In addition, radioactivity is contained in the primary and secondary coolant before the accident and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident. The St. Lucie Unit 1 AST dose analysis methodology is presented in Reference 117.

15.3.4.3.3.2 Compliance with RG 1.183 Regulatory Positions

The revised Locked Rotor dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," as discussed below:

The Locked Rotor dose consequence analysis followed the guidance provided in RG 1.183, Appendix G, "Assumptions for Evaluating the Radiological Consequences of a PWR Locked Rotor Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 15.4.1-1e. The inventory provided in Table 15.4.1-1e is adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The fraction of fission product inventory in the gap available for release due to fuel breach is consistent with Table 3 of RG 1.183. Gap release fractions have also been increased to account for high burnup fuel.
2. Regulatory Position 2 - Fuel damage is assumed for this event.
3. Regulatory Position 3 - Activity released from the damaged fuel is assumed to mix instantaneously and homogeneously throughout the primary coolant.
4. Regulatory Position 4 - The chemical form of radioiodine released from the damaged fuel is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Iodine releases from the SGs to the environment are assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to equilibrium iodine concentrations in the RCS and secondary system.
5. Regulatory Position 5.1 - The primary-to-secondary leak rate is apportioned between the SGs as specified by Technical Specification 5.5.6 (0.5 gpm total, 0.25 gpm to any one SG). Thus, the tube leakage is apportioned equally between the two SGs.
6. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is based upon RCS conditions, consistent with the plant design basis.
7. Regulatory Position 5.3 - The primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F.
8. Regulatory Position 5.4 - The analysis assumes a coincident loss of offsite power in the evaluation of fission products released from the secondary system.
9. Regulatory Position 5.5 - All noble gas radionuclides released from the primary system are assumed released to the environment without reduction or mitigation.
10. Regulatory Position 5.6 - Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows:
 - Appendix E, Regulatory Position 5.5.1 - Both steam generators are used for plant cooldown. A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncover is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.
 - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG, enter the steam space, and be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited.
 - Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage that does not flash is assumed to mix with the bulk water.

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299
478

- Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SG. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.
- Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncover in the SGs is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 1. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water of the SG into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator. A conservative uncover time of 60 minutes was assumed in the analysis.

15.3.4.3.3.3 Other Assumptions

1. RG 1.183, Section 3.6 - The assumed amount of fuel damage caused by the non-LOCA events is analyzed to determine the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and to determine the fraction of fuel elements for which fuel clad is breached. This analysis assumes DNB as the fuel damage criterion for estimating fuel damage for the purpose of establishing radioactivity releases. For the Locked Rotor event, Table 3 of RG 1.183 specifies noble gas, alkali metal, and iodine fuel gap release fractions for the breached fuel.
2. The initial RCS activity is assumed to be at the Technical Specification 3.4.16 limit of 1.0 $\mu\text{Ci/gm}$ Dose Equivalent I-131 and 518.9 $\mu\text{Ci/gm}$ DE XE-133 gross activity. The initial SG activity is assumed to be at the Technical Specification 3.7.16 limit of 0.1 $\mu\text{Ci/gm}$ Dose Equivalent I-131.
3. The steam mass release rates for the SGs are provided in Table 15.3.4-5.
4. The RCS fluid density used to convert the primary-to-secondary leakage from a volumetric flowrate to a mass flow rate is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. The high initial cooldown rate conservatively maximizes the fluid density. The SG tube leakage mass flow rate is provided in Table 15.3.4-6.
5. This RCS mass is assumed to remain constant throughout the event.
6. For the purposes of determining the iodine concentrations, the SG mass is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore secondary water level above the top of the tubes within a conservative time of one hour following a reactor trip.
7. This analysis assumes that the DNB fuel damage is limited to 19% breached fuel assemblies.

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15.3.4.3.3.4 Methodology

Input assumptions used in the dose consequence analysis of the Locked Rotor event are provided in Table 15.3.4-4. This event is caused by an instantaneous seizure of a primary reactor coolant pump rotor. Flow through the affected loop is rapidly reduced, causing a reactor trip due to a low primary loop flow signal. Following the reactor trip, the heat stored in the fuel rods continues to be transferred to the reactor coolant. Because of the reduced core flow, the coolant temperatures will rise. The rapid rise in primary system temperatures during the initial phase of the transient results in a reduction in the initial DNB margin and fuel damage.

For the purpose of this dose assessment, a total of 19% of the fuel assemblies are assumed to experience DNB. The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant. The source term is based upon release fractions from Appendix G of RG 1.183 with a radial peaking factor of 1.65. Primary coolant is released to the SGs as a result of postulated primary-to-secondary leakage. Activity is released to the atmosphere via steaming from the steam generator ADVs until the RCS is cooled to 212°F. These release assumptions are consistent with the requirements of RG 1.183.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 460 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 50-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 460 cfm of unfiltered inleakage, and 1760 cfm of filtered recirculation flow.

At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of up to 504 cfm of filtered makeup flow, 460 cfm of unfiltered inleakage, and 1256 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% elemental iodine, and 95% organic iodine.

15.3.4.3.3.5 Radiological Consequences

The atmospheric dispersion factors (χ/Q_s) used for this event for the Control Room dose are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake.

Releases are assumed to occur from the ADV that produces the most limiting χ/Q_s . When the Control Room Ventilation System is in normal mode, the most limiting χ/Q corresponds to the worst air intake to the control room. When the ventilation system is isolated, the limiting χ/Q corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. Development of control room atmospheric dispersion factors is discussed in Appendix 2J. These χ/Q_s applied in the Seized Rotor analysis are summarized in Table 15.3.4-7.

The EAB and LPZ dose consequences are determined using the χ/Q factors provided in Appendix 2I for the appropriate time intervals. For the EAB dose analysis, the χ/Q factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period.

The radiological consequences of the Locked Rotor event are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 15.3.4-8, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

The calculation of record for this analysis is in Section 15.5, Reference (8).

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TABLE 15.3.4-1

INITIAL CONDITIONS AND BIASING FOR REACTOR COOLANT PUMP ROTOR SEIZURE EVENT

<u>Parameter</u>	<u>Value</u>
Core Power	3,020 MWt + 0.3%
Core Inlet Temperature	551°F
RCS Flow Rate	375,000 gpm
Pressurizer Pressure	2,250 psia
Scram Reactivity	5000 pcm (bounding HFP minimum)
Moderator Temperature Coefficient	+2 pcm/°F
Doppler Reactivity Coefficient	-0.80 pcm/°F
Gap Conductance	Bounding BOC value
Pressurizer PORV	Available
Pressurizer Spray	Available
Pressurizer Heaters	Disabled
Low RCS Flow RPS Trip	Credited
Pressurizer Safety Valve Setpoint	2,500 psia + 3%
SG Tube Plugging	10%

TABLE 15.3.4-2

SEQUENCE OF EVENTS FOR REACTOR COOLANT PUMP ROTOR SEIZURE EVENT

<u>Event</u>	<u>Time</u>
Seizure of Pump 1A	0.0
Low RCS flow trip setpoint reached	0.168
Reactor scram (including trip response delay)	1.193
CEA insertion begins	1.693
MDNBR occurs	3.1

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TABLE 15.3.4-4

Reactor Coolant Pump Shaft Seizure (Locked Rotor) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3030 MW _{th} (3020 + 0.3%)
Core Average Fuel Burnup	49,000 MWD/MTU
Fuel Enrichment	1.5 – 5.0w/o
Maximum Radial Peaking Factor	1.65
Percent of Fuel Rods in DNB	19%
Core Fission Product Inventory	Table 15.4.1-1e
Initial RCS Equilibrium Activity	1.0 μCi/gm DE I-131 and 518.9 μCi/gm DE XE-133 gross activity (Table 15.4.1-9)
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131 (Table 15.4.6-8)
Release Fraction from Breached Fuel	RG 1.183, Section 3.2
Steam Generator Tube Leakage	0.5 gpm (Table 15.3.4-6)
Time to Terminate SG Tube Leakage	12.4 hours
Secondary Side Mass Releases to Environment	Table 15.3.4-5
SG Tube Uncovery Following Reactor Trip Time to tube recovery Flashing Fraction	1 hour 5 %
Steam Generator Secondary Side Partition Coefficient	Flashed tube flow – none Non-flashed tube flow – 100
Time to Reach 212°F and Terminate Steam Release	12.4 hours
RCS Mass	406,715 lb _m Minimum mass used for fuel failure dose contribution to maximize SG tube leakage activity.
SG Secondary Side Mass	minimum – 120,724 lb _m (per SG) maximum – 226,800 lb _m (per SG) Minimum used for primary-to-secondary leakage to maximize secondary nuclide concentration. Maximum used for initial secondary inventory release to maximize secondary side dose contribution.
Atmospheric Dispersion Factors Offsite Onsite	Appendix 2I Tables 15.3.4-7
Control Room Ventilation System Time of Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	50 seconds 1.5 hours 460 cfm
Breathing Rates Offsite Onsite	RG 1.183 Section 4.1.3 RG 1.183 Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

TABLE 15.3.4-4a

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TABLE 15.3.4-5

LOCKED ROTOR STEAM RELEASE RATE

Time (hours)	Intact Steam Generator SteamRelease Flow Rate (lb _m /min)
0	5486
0.50	2821
0.75	2821
1.39	2821
2.00	2846
4.00	2846
6.00	2846
8.00	2846
10.50	2846
12.40	0.00

TABLE 15.3.4-6

LOCKED ROTOR SG TUBE LEAKAGE

Time	Total SG Tube Leakage Flow Rate (lb _m /min)
0	3.103
0.50	3.361
0.75	3.428
1.00	3.536
1.39	3.565
2.00	3.657
4.00	3.756
8.00	3.945
10.50	4.012
12.40	0.000

TABLE 15.3.4-7

CONTROL ROOM χ/Q_s

Time (hours)	χ/Q (sec/m ³)
0	6.30E-3
0.013889	2.84E-3
1.5	1.62E-3
2.0	1.32E-3
8.0	5.06E-4
24.0	3.88E-4
96.0	3.30E-4
720.0	3.30E-4

TABLE 15.3.4-8

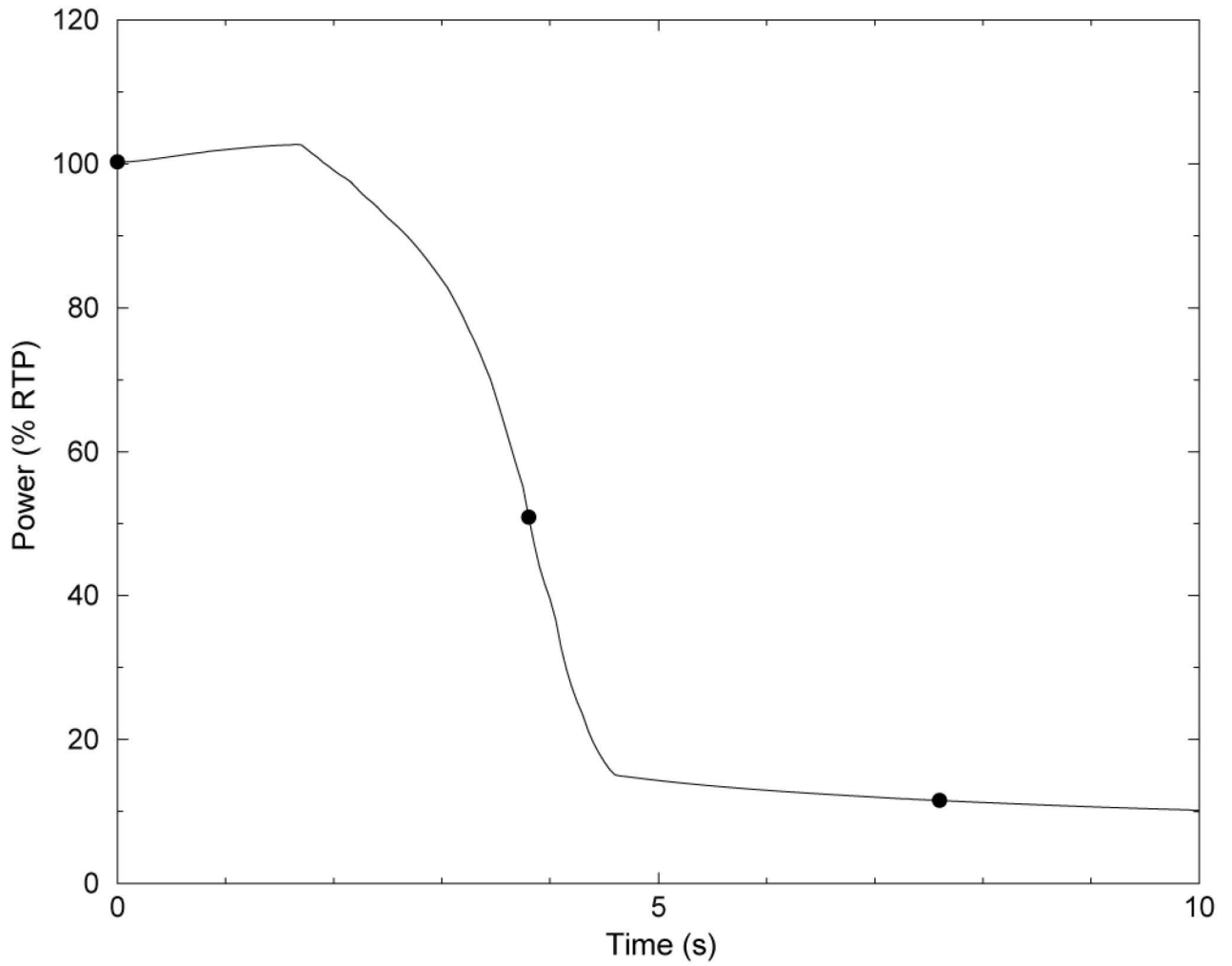
LOCKED ROTOR DOSE CONSEQUENCES

Case	EAB Dose ⁽¹⁾ (REM TEDE)	LPZ Dose ⁽²⁾ (REMTED)	Control Room Dose ⁽²⁾ (REM TEDE)
Locked Rotor	0.37	0.87	4.38
Acceptance Criteria	2.5	2.5	5

⁽¹⁾ Worst 2-hour dose

⁽²⁾ Integrated 30-day dose

Reactor Power – Seized Rotor Event



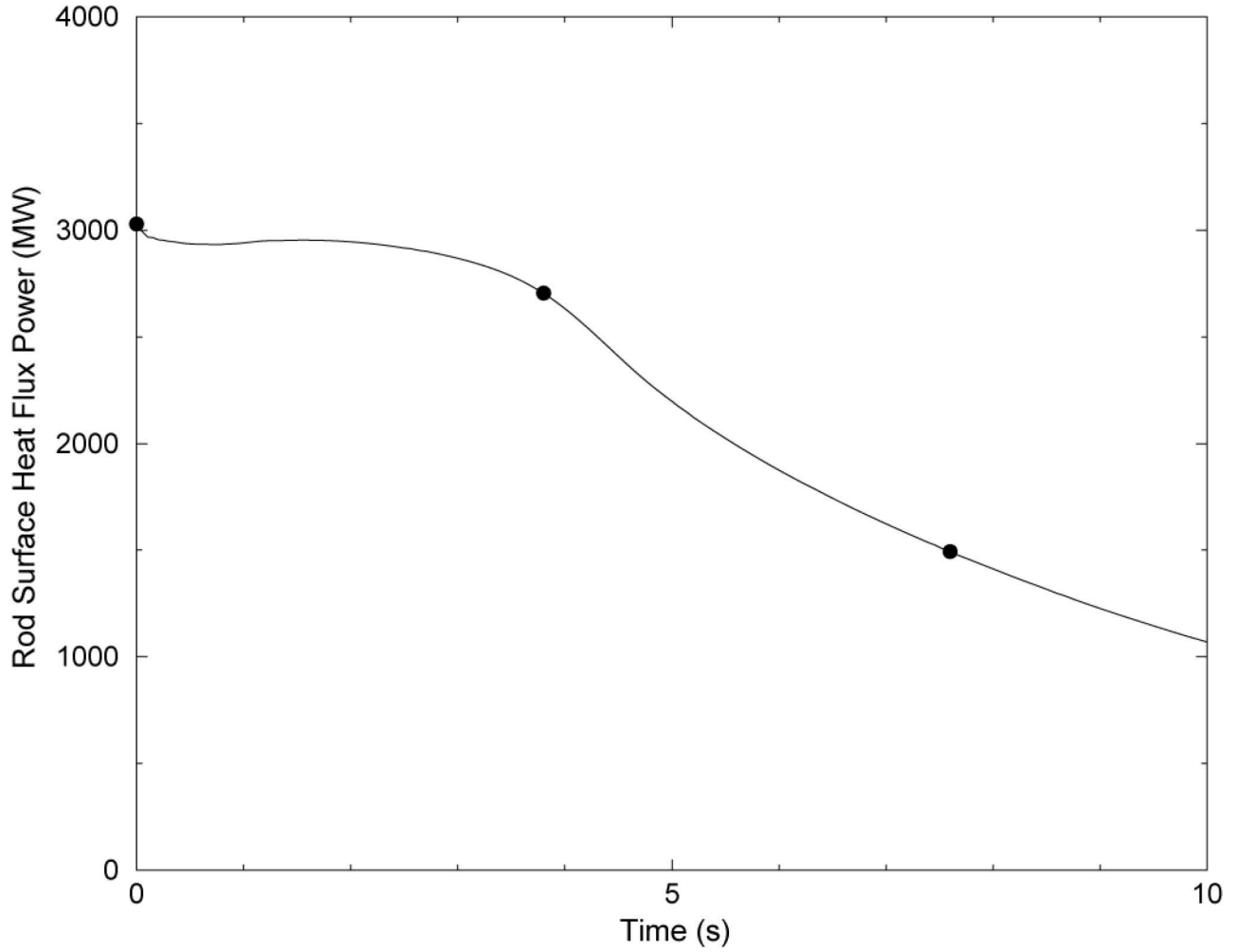
Amendment No. 26 (11/13)

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ST. LUCIE PLANT UNIT 1**

**REACTOR POWER
SEIZED ROTOR EVENT**

FIGURE 15.3.4-1

Total Core Heat Flux Power – Seized Rotor Event



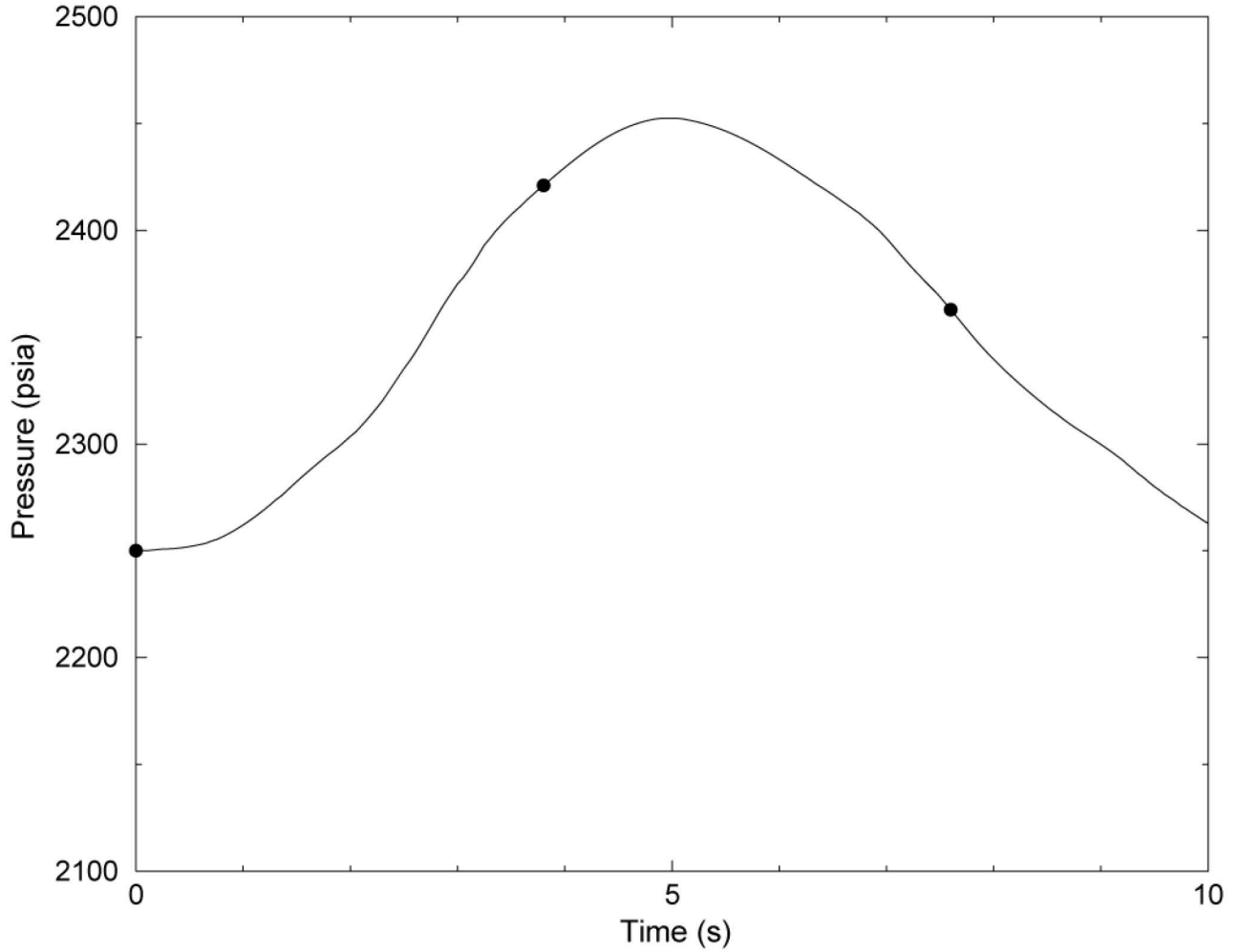
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**TOTAL CORE HEAT FLUX POWER
SEIZED ROTOR EVENT**

FIGURE 15.3.4-2

Pressurizer Pressure – Seized Rotor Event



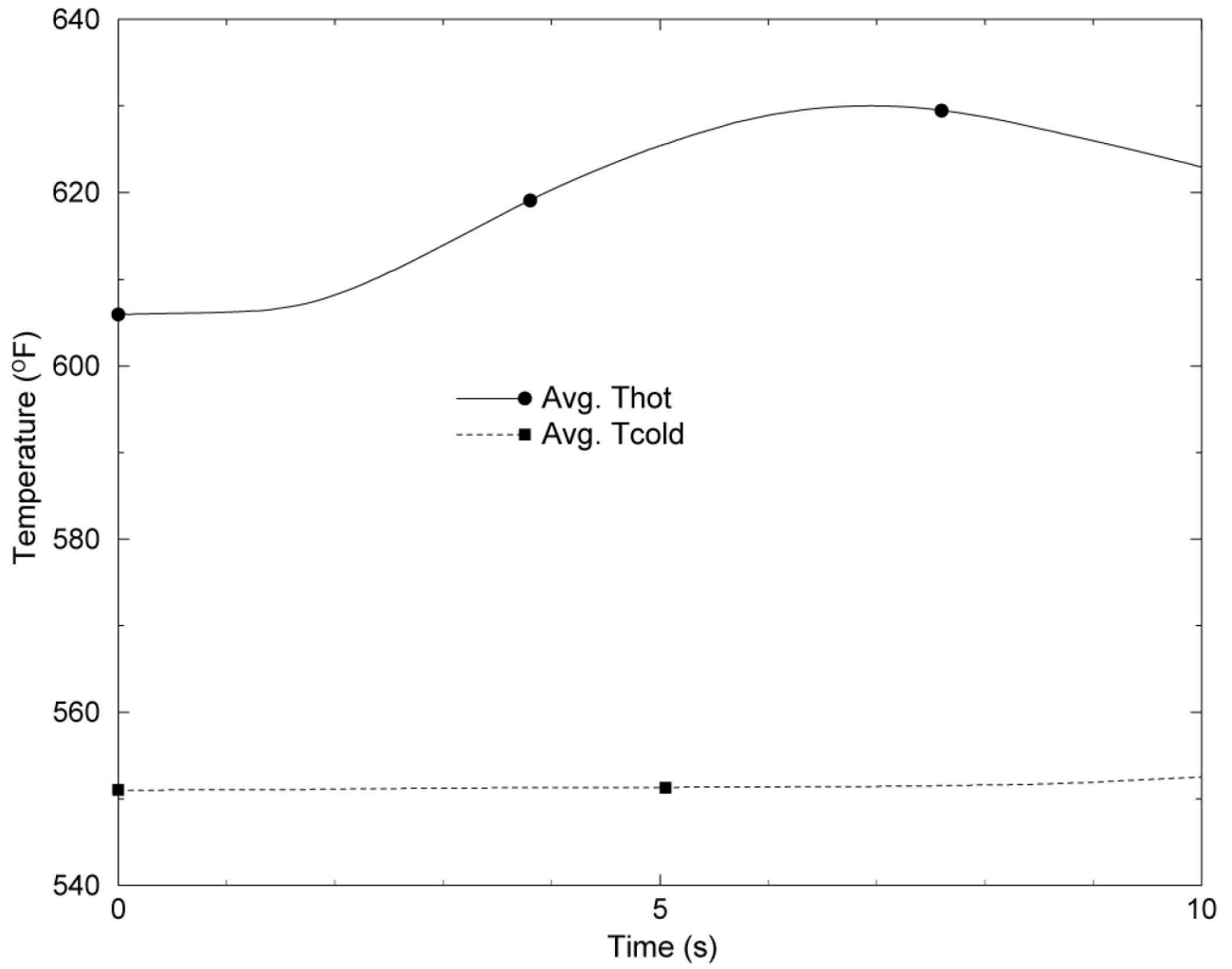
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**PRESSURIZER PRESSURE
SEIZED ROTOR EVENT**

FIGURE 15.3.4-3

RCS Loop Temperatures – Seized Rotor Event



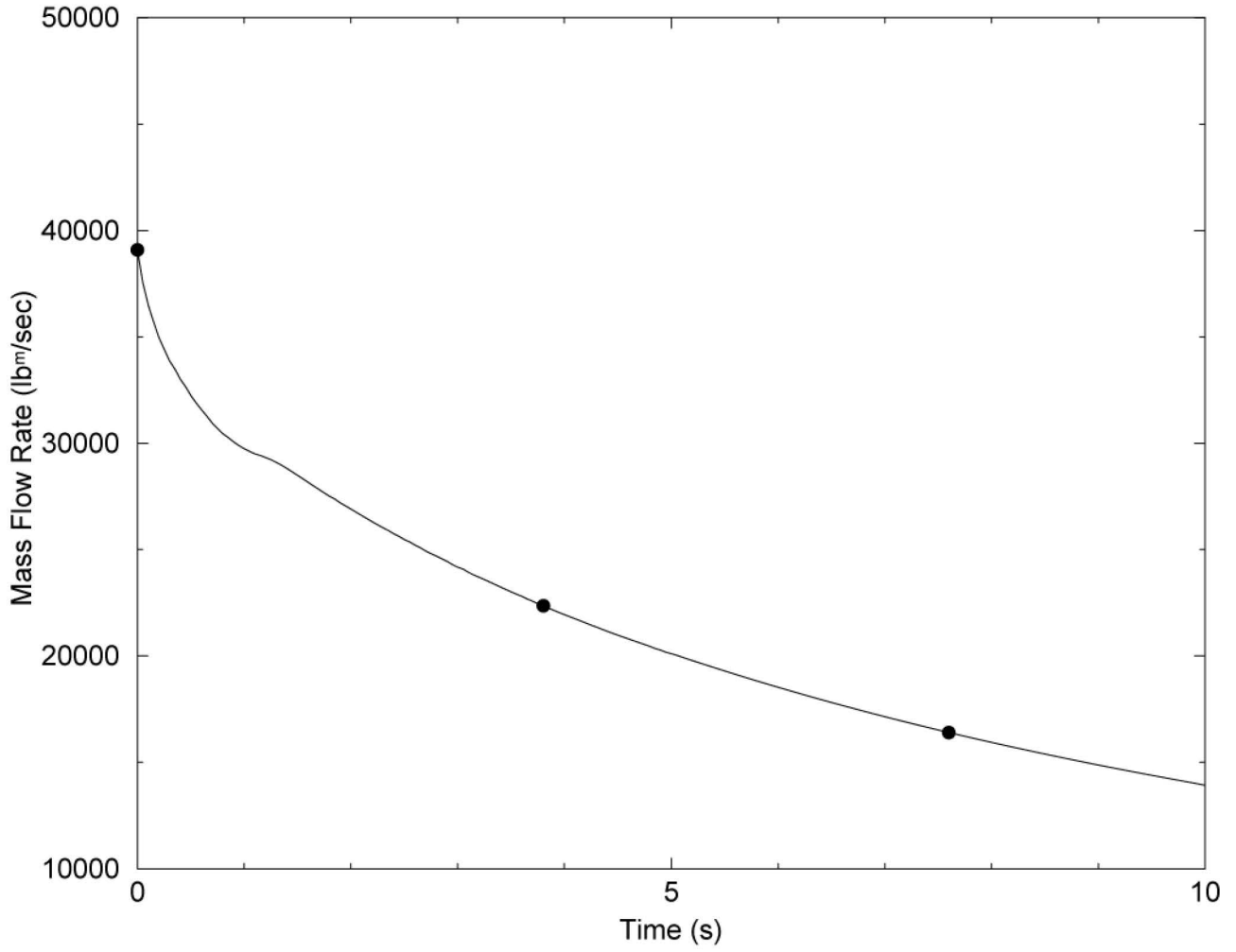
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**RCS LOOP TEMPERATURES
SEIZED ROTOR EVENT**

FIGURE 15.3.4-4

RCS Total Loop Flow Rate– Seized Rotor Event



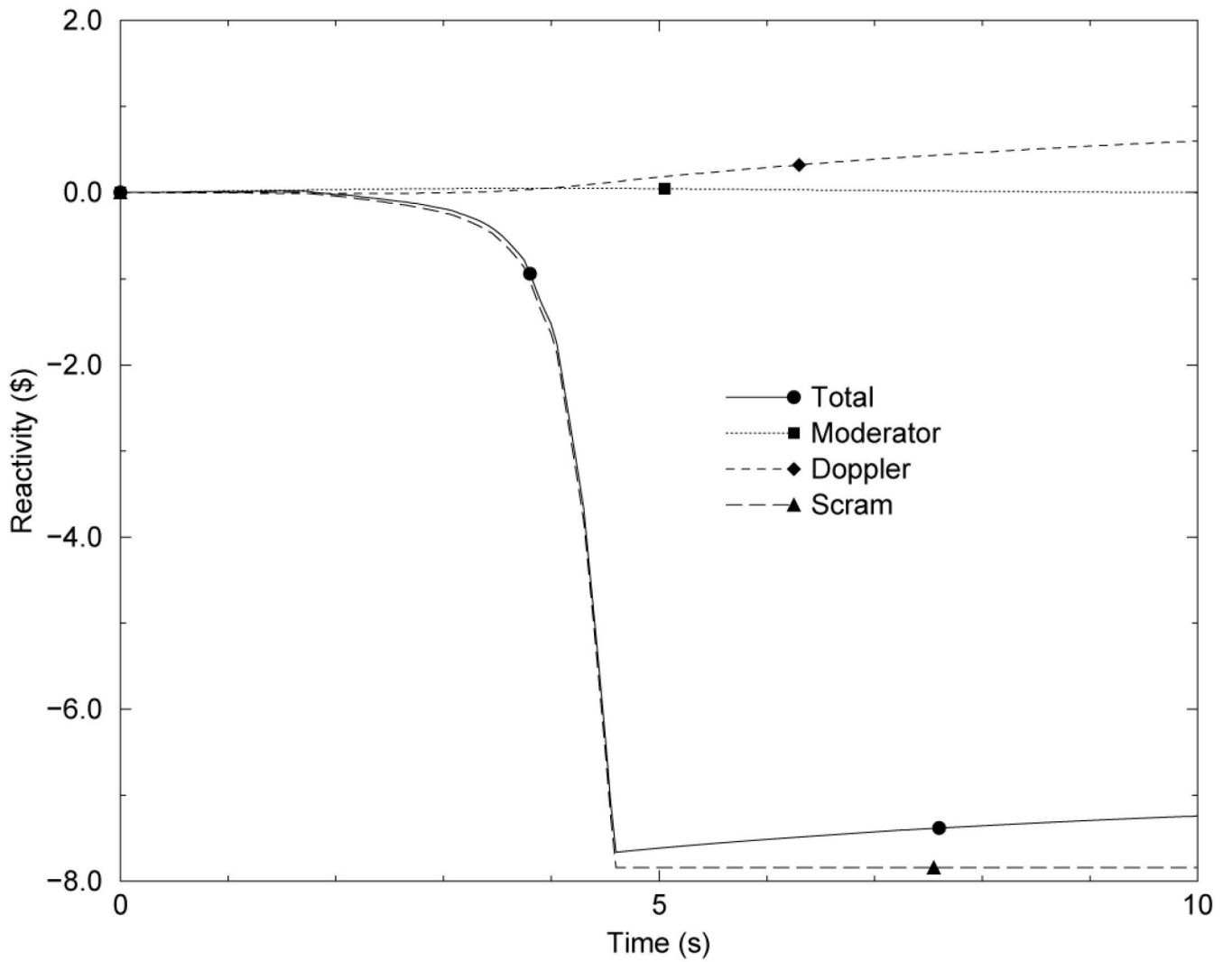
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**RCS TOTAL LOOP FLOW RATE
SEIZED ROTOR EVENT**

FIGURE 15.3.4-5

Reactivity Feedback – Seized Rotor Event



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REACTIVITY FEEDBACK
SEIZED ROTOR EVENT

FIGURE 15.3.4-6

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**SEIZED ROTOR (DNB) EVENT
COLD LEG FLOWS**

FIGURE 15.3.4-7

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**SEIZED ROTOR (DNB) EVENT
PRESSURES**

FIGURE 15.3.4-8

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**SEIZED ROTOR (DNB) EVENT
WATER LEVELS**

FIGURE 15.3.4-9

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**SEIZED ROTOR (DNB) EVENT
DNBR**

FIGURE 15.3.4-10

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**SEIZED ROTOR (DNB) EVENT
REACTIVITY**

FIGURE 15.3.4-11

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**SEIZED ROTOR (PRESSURE) EVENT
POWER, HEAT, FLUX AND FLOW**

FIGURE 15.3.4-12

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**SEIZED ROTOR (PRESSURE) EVENT
S. G. FLOWS**

FIGURE 15.3.4-13

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**SEIZED ROTOR (PRESSURE) EVENT
FUEL TEMPERATURE**

FIGURE 15.3.4-14

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**SEIZED ROTOR (PRESSURE) EVENT
CORE TEMPERATURES**

FIGURE 15.3.4-15

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Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
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**SEIZED ROTOR (PRESSURE) EVENT
LOOP TEMPERATURE DIFFERENCES**

FIGURE 15.3.4-16

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**SEIZED ROTOR (PRESSURE) EVENT
AVERAGE TEMPERATURES**

FIGURE 15.3.4-17

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**FLORIDA POWER & LIGHT COMPANY
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**SEIZED ROTOR (PRESSURE) EVENT
COLD LEG FLOWS**

FIGURE 15.3.4-18

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Amendment No. 26 (11/13)

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**SEIZED ROTOR (PRESSURE) EVENT
PRESSURES**

FIGURE 15.3.4-19

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Amendment No. 26 (11/13)

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**SEIZED ROTOR (PRESSURE) EVENT
WATER LEVELS**

FIGURE 15.3.4-20

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**SEIZED ROTOR (PRESSURE) EVENT
DNBR**

FIGURE 15.3.4-21

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**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**SEIZED ROTOR (PRESSURE) EVENT
REACTIVITY**

FIGURE 15.3.4-22

15.4 POSTULATED ACCIDENTS (CLASS 3 ACCIDENTS)

The following events in this class had been reanalyzed for the extended power uprate (3020 MWt) to ensure acceptable consequences.

15.4.1 Major Reactor Coolant System Pipe Break (Loss of Coolant Accident)

15.4.4 Steam Generator Tube Failure

15.4.5 Control Element Assembly Ejection

15.4.6 Steam Line Break Accident

The other events have not changed from the original FSAR submittal. Reanalysis has not been required for these events because:

- a) no significant parameters have changed, or
- b) parameter changes have occurred but have no significance to results, or
- c) the uprated core power level of 3020 MWth has insignificant effect on the results.

15.4.1 MAJOR REACTOR COOLANT SYSTEM PIPE BREAK (LOSS OF COOLANT ACCIDENT)

For stretch power operation (2700 MWth) this event was analyzed by Combustion Engineering and is described in Chapter 6. A radiological dose reassessment of the LOCA was not required since the original analysis was performed at an assumed power level of 2700 MWth. The original assessment is presented in Section 15.4.1.8.

15.4.1.1 Identification of Causes*

A major loss of coolant accident (LOCA) is defined as a break in the reactor coolant pressure boundary having an area greater than 0.5 ft². Such a break and the consequent coolant loss would result in loss of the normal mechanism for removing heat from the reactor core.

Because of the extreme care taken in design and fabrication of the plant, and because of the periodic testing and in-service inspection required, the probability of a major LOCA is considered to be extremely low. Nevertheless, because of the potential consequences, several important systems identified as engineered safety features have been provided to prevent the clad and fuel from melting, to limit chemical reactions, and to protect the health and safety of the public. These systems, Section 6, are the safety injection system (emergency core cooling system or ECCS) Section 6.3; the containment heat removal systems (containment spray and containment cooling) Section 6.2.2; the containment isolation system, Section 6.2.4; the shield building ventilation system, Section 6.2.3; and hydrogen control system, Section 6.2.5.

The analysis of the LOCA is discussed in Chapter 6. The initial conditions, assumptions, and a step by step sequence of events are an integral part of that discussion. The response of the safety injection system is described in Section 6.3; the response of the remaining containment related engineered safety features is described in Section 6.2

*CE-Analysis prior to Cycle 6

15.4.1.2 Reload Safety Analysis

15.4.1.2.1 Introduction

This section describes and provides results from a Realistic Large Break LOCA (RLBLOCA) analysis for the Saint Lucie Nuclear Plant Unit I Extended Power Uprate. The uprated reactor core power for the St. Lucie Unit I RLBLOCA is 3029.1 MWt. This value represents the 10% power uprate and 1.7% measurement uncertainty recapture (MUR) relative to the previously rated thermal power of 2700 MWt plus 0.3% power measurement uncertainty.

The RLBLOCA analysis was performed with a version of S-RELAP5 that limits the contribution of the Forslund-Rohsenow model to no more than 15 percent of the total heat transfer at and above a void fraction of 0.9. This may result in a slight increase in PCT results when compared to previous analyses for similar plants.

In concurrence with the NRC's interpretation of GDC 35, a set of 59 cases was run with a Loss of Offsite Power (LOOP) assumption and a second set was run with No-LOOP assumption to search for the limiting PCT case. The results from both case sets are shown in Figure 15.4.1-27.

15.4.1.2.2 Summary of Parameters

The summary of major parameters for the limiting PCT case is shown in Table 15.4.1-1c.

15.4.1.2.3 Large Break LOCA Analysis

The purpose of the analysis is to verify typical technical specification peaking factor limits and the adequacy of the ECCS by demonstrating that the following 10 CFR 50.46(b) criteria in Reference 1 are met:

- (1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- (2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.
- (3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel excluding the cladding surrounding the plenum volume were to react.
- (4) The calculated changes in core geometry shall be such that the core remains amenable to cooling.
- (5) Long-term cooling is not addressed in this calculation.

The analysis did not evaluate core coolability due to seismic events, nor did it consider the 10 CFR 50.46(b) long-term cooling criterion.

15.4.1.2.3.1 LBLOCA Event Description

A LBLOCA is initiated by a postulated large rupture of the reactor coolant system (RCS) primary piping. Based on deterministic studies, the worst break location is in the cold leg piping between the reactor coolant pump and the reactor vessel for the RCS loop containing the pressurizer. The break initiates a

rapid depressurization of the RCS. A reactor trip signal is initiated when the low pressurizer pressure trip setpoint is reached; however, reactor trip is conservatively neglected in the LBLOCA analysis. The reactor is shut down by coolant voiding in the core.

The plant is assumed to be operating normally at full power prior to the accident. The cold leg break is assumed to open instantaneously. For this break, a rapid depressurization occurs, along with a core flow stagnation and reversal. This causes the fuel rods to experience departure from nucleate boiling (DNB). Subsequently, the limiting fuel rods are cooled by film convection to steam. The coolant voiding creates a strong negative reactivity effect and core criticality ends. As heat transfer from the fuel rods is reduced, the cladding temperature increases.

Coolant in all regions of the RCS begins to flash. At the break plane, the loss of subcooling in the coolant results in substantially reduced break flow. This reduces the depressurization rate, and leads to a period of positive core flow or reduced downflow as the RCPs in the intact loops continue to supply water to the RV (in No-LOOP condition). Cladding temperatures may be reduced and some portions of the core may rewet during this period. The positive core flow or reduced downflow period ends as two phase conditions occur in the RCPs, reducing their effectiveness. Once again, the core flow reverses as most of the vessel mass flows out through the broken cold leg.

Mitigation of the LBLOCA begins when the safety injection actuation signal (SIAS) is issued. This signal is initiated by either high containment pressure or low pressurizer pressure. Regulations require that a worst single failure be considered. This single failure has been determined to be the loss of one ECCS pumped injection train. The Framatome (formerly AREVA) Realistic Large Break LOCA (RLBLOCA) methodology conservatively assumes an on-time start and normal lineups of the containment spray to conservatively reduce containment pressure and increase break flow. Hence, the analysis assumes that the loss of a diesel generator, which takes one train of ECCS pumped injection out. LPSI injects into the broken loop and one intact loop, HPSI injects into all four loops, and all containment spray pumps are operating.

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When the RCS pressure falls below the SIT (Safety Injection Tank) pressure, fluid from the SITs is injected into the cold legs. In the early delivery of SIT water, high pressure and high break flow will drive some of this fluid to bypass the core. During this bypass period, core heat transfer remains poor and fuel rod cladding temperatures increase. As RCS and containment pressures equilibrate, ECCS water begins to fill the lower plenum and eventually the lower portions of the core; thus, core heat transfer improves and cladding temperatures decrease.

Eventually, the relatively large volume of SIT water is exhausted and core recovery continues relying solely on pumped ECCS injection. As the SITs empty, the nitrogen gas used to pressurize the SITs exits through the break. This gas release may result in a short period of improved core heat transfer as the nitrogen gas displaces water in the downcomer. After the nitrogen gas has been expelled, the ECCS temporarily may not be able to sustain full core cooling because of the core decay heat and the higher steam temperatures created by quenching in the lower portions of the core. Peak fuel rod cladding temperatures may increase for a short period until more energy is removed from the core by the HPSI and LPSI while the decay heat continues to fall. Steam generated from fuel rod rewet will entrain liquid and pass through the core, vessel upper plenum, the hot legs, the steam generators, and the reactor coolant pumps before it is vented out the break. Some steam flow to the upper head and pass through the spray nozzles, which provide a vent path to the break. The resistance of this flow path to the steam flow is balanced by the driving force of water filling the downcomer. This resistance may act to retard the progression of the core reflood and postpone core-wide cooling. Eventually (within a few minutes of the accident), the core reflood will progress sufficiently to ensure core-wide cooling. Full core quench occurs within a few minutes after core-wide cooling. Long term cooling is then sustained with LPSI pumped injection system.

15.4.1.2.3.2 Description of Analytical Models

The RLBLOCA methodology is documented in EMF-2103 Realistic Large Break LOCA Methodology (Reference 2). The methodology follows the Code Scaling, Applicability, and Uncertainty (CSAU) evaluation approach (Reference 3). This method outlines an approach for defining and qualifying a best estimate thermal hydraulic code and quantifies the uncertainties in a LOCA analysis.

The RLBLOCA methodology employs the following computer codes:

- RODEX3A for computation of the initial fuel stored energy, fission gas release, and fuel cladding gap conductance.
- S-RELAP5 for the system calculation (includes ICECON for containment response).
- AUTORLBLOCA for generation of ranged parameter values, transient input, transient runs, and general output documentation.

NRC Information Notice 2009-23 (Reference 4) describes an issue concerning the ability of thermal-mechanical fuel modeling codes (such as RODEX3A) to accurately predict the exposure-dependent degradation of fuel thermal conductivity. To address this issue, the analytical model applies a conservative bias as an adjustment to the calculation of fuel thermal conductivity and centerline temperature. In addition, to better follow the fuel throughout its operational life, the analytical method has been updated to specifically model both first (i.e., fresh) and second cycle (i.e., once-burned) fuel rods. The third cycle fuel does not retain sufficient energy to achieve either significant cladding temperatures or cladding oxidation and so is not explicitly modeled. This approach has been approved by the NRC for St. Lucie Unit 1.

The governing two fluid (plus non-condensibles) model with conservation equations for mass, energy, and momentum transfer is used. The reactor core is modeled in S-RELAP5 with heat generation rates determined from reactor kinetics equations (point kinetics) with reactivity feedback, and with actinide and decay heating. The two fluid formulation uses a separate set of conservation equations and constitutive relations for each phase. The effects of one phase on the other are accounted for by interfacial friction, and heat and mass transfer interaction terms in the equations. The conservation equations have the same form for each phase; only the constitutive relations and physical properties differ.

The modeling of plant components is performed by following guidelines developed to ensure accurate accounting for physical dimensions and that the dominant phenomena expected during the LBLOCA event are captured. The basic building blocks for modeling are hydraulic volumes for fluid paths and heat structures for heat transfer. In addition, special purpose components exist to represent specific components such as the RCPs or the steam generator separators. All geometries are modeled at the resolution necessary to best resolve the flow field and the phenomena being modeled within practical computational limitations.

A steady state conditions is established with all loops intact. The input parameters and initial conditions for this steady state calculation are chosen to reflect plant technical specifications or to match measured data. Following the establishment of an acceptable steady state condition, the transient calculation is initiated by introducing a break into the loop containing the pressurizer. The evolution of the transient through blowdown, refill and reflood is computed continuously using S-RELAP5. Containment pressure is also calculated by S-RELAP5 using containment models derived from ICECON (Reference 5), which

is based on the CONTEMPT-LT code (Reference 6) and has been updated for modeling ice condenser containments. The methods used in the application of S-RELAP5 to the LBLOCA are described in Reference 2.

15.4.1.2.3.3 Plant Description and Summary of Analysis Parameters

St. Lucie Unit 1 is a CE-designed PWR, which has two hot legs, two U-tube steam generators, and four cold legs with one RCP in each cold leg. The plant uses a large dry containment. The RCS includes one Pressurizer connected to a hot leg. The core contains 217 thermal hydraulic compatible Framatome (formerly AREVA) HTP 14X14 fuel assemblies with 2, 4, 6 and 8 w/o gadolinia pins. The ECCS includes one HPSI, one LPSI and one SIT injection path per RCS loop. The break is modeled in the same loop as the pressurizer, as directed by the RLBLOCA methodology. The RLBLOCA transients are of sufficiently short duration that the switchover to sump cooling water for ECCS pumped injection need not be considered.

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The S-RELAP5 model explicitly describes the RCS, RV, pressurizer, and ECCS. The ECCS includes a SIT path and a LPSI/HPSJ path per RCS loop. The HPSI and LPSI feed into a common header that connects to each cold leg pipe downstream of the RCP discharge. The ECCS pumped injection is modeled as a table of flow versus backpressure. Applying the worst single failure of one emergency diesel generator affects the ECCS pumped injection systems available, injection location, and pumped ECCS flow. A table of flow versus backpressure also describes the secondary side steam generator that is instantaneously isolated (closed MSIV and feedwater trip) at the time of the break. A symmetric steam generator tube plugging level of 10 percent per steam generator was assumed.

As described in the Framatome (formerly AREVA) RLBLOCA methodology, many parameters associated with LBLOCA phenomenological uncertainties and plant operation ranges are sampled. A summary of those parameters is given in Table 15.4.1-1. The LBLOCA phenomenological uncertainties are provided in Reference 2. Values for process or operational parameters, including ranges of sampled process parameters, and fuel design parameters used in the analysis are given in Table 15.4.1-1a. Plant data are analyzed to develop uncertainties for the process parameters sampled in the analysis. Table 15.4.1-1 b presents a summary of the uncertainties used in the analysis. Where applicable, the sampled parameter ranges are based on technical specification limits or supporting plant calculations that provide more bounding values.

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For the Framatome (formerly AREVA) RLBLOCA EM, dominant containment parameters, as well as NSSS parameters, were established via a PIRT process. Other model inputs are generally taken as nominal or conservatively biased. The PIRT outcome yielded two important (relative to PCT) containment parameters--containment pressure and temperature. As noted in Table 15.4.1-1b, containment temperature is a sampled parameter. Containment pressure response is indirectly ranged by sampling the containment volume (Table 15.4.1-1b). Containment heat sink data is given in Table 15.4.1-1f. The containment initial conditions and boundary conditions are given in Table 15.4.1-1g. The building spray is modeled at maximum heat removal capacity. All spray flow is delivered to the containment.

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15.4.1.2.3.4 LBLOCA Results

Two case sets of 59 transient calculations were performed by sampling the parameters listed in Table 15.4.1-1. For each case set, a PCT was calculated for a UO₂ rod and for Gadolinia-bearing rods with concentrations of 2, 4, 6 and 8 w/o Gd₂O₃. Both fresh and once-burnt fuel are considered. The limiting case set, containing the highest PCT, corresponds to that with no offsite power available. A limiting PCT of 1788°F appears in Case 23 for a UO₂ rod in a fresh bundle. The major parameters for the limiting transient are summarized in Table 15.4.1-1c. Table 15.4.1-1d summarizes the results of the limiting case. The fraction of total hydrogen generated was not directly calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is well below the 1 percent limit.

The case results, event times and analysis plots for the limiting PCT case are shown in Table 15.4.1-1 d, Table 15.4.1-1e, and in Figure 15.4.1-6 through Figure 15.4.1-26. Figure 15.4.1-1 shows linear scatter plots of the key parameters sampled from the 59 calculations. Parameter labels appear to the left of each individual plot. These figures show the parameter ranges used in the analysis. Figure 15.4.1-2 and Figure 15.4.1-3 show PCT versus PCT time scatter plot and PCT versus break size scatter plot from the 59 calculations, respectively. Figure 15.4.1-4 and Figure 15.4.1-5 show the maximum oxidation and total oxidation versus PCT scatter plots from the 59 calculations, respectively. Key parameters for the limiting PCT case are shown in Figure 15.4.1-6 through Figure 15.4.1-16. Additional information from the limiting PCT case is presented in Figure 15.4.1-17 through Figure 15.4.1-26. Figure 15.4.1-6 is the plot of PCT (independent of elevation) versus time for the limiting case; this figure clearly indicates that the transient exhibits a sustained and stable quench. A comparison of PCT results between the LOOP and no-LOOP case sets is shown in Figure 15.4.1-27. As seen in Figure 15.4.1-27, the peak PCT is from a limiting LOOP case.

15.4.1.2.4 Conclusions

A RLBLOCA analysis was performed for the St Lucie Nuclear Plant Unit 1 using NRC-approved Framatome (formerly AREVA) RLBLOCA methods (Reference 2). Analysis results show that a LOOP case is limiting and has a PCT of 1788^{°F(1)} and a maximum oxidation thickness and hydrogen generation that fall well within regulatory requirements.

The analysis supports operation at a nominal power level of 3029.1 MWt (including 0.3% uncertainty), a steam generator tube plugging level of up to 10 percent in all steam generators, a total LHGR of 15.0 kW/ft, a total peaking factor (F_Q) up to a value of 2.161, and a nuclear enthalpy rise factor ($F_{\Delta H}$) up to a value of 1.810 (including 6% measurement uncertainty and 3.5% allowance for control rod insertion effect) with no axial or bumup dependent power peaking limit and peak rod average exposures of up to 62,000 MWd/MTU. For a large break LOCA, the three 10 CFR 50.46 (b) criteria presented in Section 15.4.1.2.3 are met and operation of St. Lucie Unit 1 with Framatome (formerly AREVA)-supplied 14x14 M5 cladding fuel is justified.

The calculation of record for this analysis is in Section 15.5, Reference (8).

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(1) The PCT changes due to evaluation model errors/changes are documented in the 10 CFR 50.46 reports. These PCT changes are not (or may not be) reflected in the PCT document here.

15.4.1.2.5 REFERENCES for Section 15.4.1.2

1. Title 10, Code of Federal Regulations, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."
2. EMF-2103(P)(A) Revision 0, Realistic Large Break LOCA Methodology, Framatome ANP, Inc., April 2003 (as supplemented by ANP-2903(P), "St. Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA Summary Report Zr-4 Fuel Cladding," Revision 1).
3. Technical Program Group, Quantifying Reactor Safety Margins, NUREG/CR-5249, EGG-2552, October 1989.
4. U.S. Nuclear Regulatory Commission, Information Notice 2009-23, ML091550527, "Nuclear Fuel Thermal Conductivity Degradation," October 8, 2009.
5. XN-CC-39 (A) Revision 1, "ICECON: A Computer Program to Calculate Containment Back Pressure for LOCA Analysis (Including Ice Condenser Plants)," Exxon Nuclear Company, October 1978.
6. Wheat, Larry L., "CONTEMPT-LT A Computer Program for Predicting Containment Pressure Temperature Response to a Loss-Of-Coolant-Accident," Aerojet Nuclear Company, TID-4500, ANCR-1219, June 1975.
7. U. S. Nuclear Regulatory Commission, NUREG-0800, Revision 3, Standard Review Plan, March 2007.

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15.4.1.5 Radiological Consequences

15.4.1.5.1 Background

This event is assumed to be caused by an abrupt failure of the main reactor coolant pipe and the ECCS fails to prevent the core from experiencing significant degradation (i.e., melting). This sequence cannot occur unless there are multiple failures, and thus goes beyond the typical design basis accident that considers a single active failure. Activity is released from the containment and from there, released to the environment by means of containment leakage and leakage from the ECCS. The St. Lucie Unit 1 AST dose analysis methodology is presented in Reference 107.

15.4.1.5.2 Compliance with RG 1.183 Regulatory Positions

The revised LOCA dose consequence analysis is consistent with the guidance provided in RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, at 100.3% of core thermal power and is provided in Table 15.4.1-1e. The core inventory release fractions for the gap release and early in-vessel damage phases of the LOCA are consistent with Regulatory Position 3.2 and Table 2 of RG 1.183.
2. Regulatory Position 2 - Per Section 6.2.6.1, the long term recirculation sump pH remains greater than 7.0. Therefore, the chemical form of the radioiodine released to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form.
3. Regulatory Position 3.1 - The activity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment. The release into the containment is assumed to terminate at the end of the early in-vessel phase.
4. Regulatory Position 3.2 - Reduction of the airborne radioactivity in the containment by natural deposition is credited. A natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 2.89 hr^{-1} . This removal is credited in both the sprayed and unsprayed regions of containment. A natural deposition removal coefficient of 0.1 hr^{-1} is assumed for all aerosols in the unsprayed region of containment as well as in the sprayed region after spray is terminated at 8 hours. No removal of organic iodine by natural deposition is assumed.
5. Regulatory Position 3.3 - Containment spray provides coverage to 86% of the containment. Therefore, the St. Lucie Unit 1 containment building atmosphere is not considered to be a single, well-mixed volume. A mixing rate of two turnovers of the unsprayed region per hour is assumed. The maximum decontamination factor (DF) for the elemental iodine spray removal coefficient is 200 based on the maximum airborne elemental iodine concentration in the containment. Based upon the conservatively assumed elemental iodine removal rate of 20 hr^{-1} , the DF of 200 is computed to occur at 2.331 hours. In addition, the particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated iodine aerosol removal rate of 6.07 hr^{-1} , the DF of 50 is conservatively computed to occur at 2.334 hours.

6. Regulatory Position 3.4 - Reduction in airborne radioactivity in the containment by filter recirculation systems is not assumed in this analysis.
7. Regulatory Position 3.5 - This position relates to suppression pool scrubbing in BWRs, which is not applicable to St. Lucie Unit No. 1.
8. Regulatory Position 3.6 - This position relates to activity retention in ice condensers, which is not applicable to St. Lucie Unit No. 1.
9. Regulatory Position 3.7 - A containment leak rate of 0.5 Vol. % per day of the containment air is assumed for the first 24 hours. After 24 hours, the containment leak rate is reduced to 0.25 Vol. % per day of the containment air.
10. Regulatory Position 3.8 - 100% of the radionuclide inventory of the RCS is released instantaneously at the beginning of the event. The containment purge flow is 500 cfm and is assumed to be isolated after 30 seconds. No filters are credited.
11. Regulatory Position 4.1 - Leakage from containment collected by the secondary containment is processed by ESF filters prior to an assumed ground level release.
12. Regulatory Position 4.2 - Leakage into the secondary containment is assumed to be released directly to the environment as a ground level release prior to drawdown of the secondary containment at 310 seconds.
13. Regulatory Position 4.3 - SBVS is credited as being capable of maintaining the Shield Building Annulus at a negative pressure with respect to the outside environment considering the effect of high windspeeds and LOCA heat effects on the annulus as described in Section 6.2. No exfiltration through the concrete wall of the Shield Building is expected to occur.
14. Regulatory Position 4.4 - No credit is taken for dilution in the secondary containment volume.
15. Regulatory Position 4.5 - 9.6% of the primary containment leakage is assumed to bypass the secondary containment. This bypass leakage is released as a ground level release without credit for filtration.
16. Regulatory Position 4.6 - The SBVS is credited as meeting the requirements of RG 1.52 and Generic Letter 99-02. The filters in the SBVS ventilation system are credited at 99% efficiency for particulates and 95% for both elemental and organic iodine.
17. Regulatory Position 5.1 - Engineered Safety Feature (ESF) systems that recirculate water outside the primary containment are assumed to leak during their intended operation. With the exception of noble gases, all fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix in the containment sump water at the time of release from the core.
18. Regulatory Position 5.2 - Leakage from the ESF system is greater than two times the value identified in UFSAR Table 15.4.1-2 for pump seals and valve stems in the ECCS area. The leakage is assumed to start at the earliest time the recirculation flow occurs in these systems and continue for the 30-day duration. Backleakage to the RWT is also considered separately as two times 1 gpm, which is the bounding value based upon RCS leakage monitoring documented in the Control Room database. RWT leakage is assumed to begin at the start of recirculation and continue for the remainder of the 30-day duration.

19. Regulatory Position 5.3 - With the exception of iodine, all radioactive materials in the recirculating liquid are assumed to be retained in the liquid phase.
20. Regulatory Position 5.4 - A flashing fraction of 7.5% was calculated based upon the sump temperature at the time of recirculation. However, consistent with Regulatory Position 5.5, the flashing fraction for ECCS leakage is assumed to be 10%. This ECCS leakage enters the Reactor Auxiliary Building. For ECCS leakage back to the RWT, the analysis demonstrates that the temperature of the leaked fluid will cool below 212°F prior to release into the tank.
21. Regulatory Position 5.5 - The amount of iodine that becomes airborne is conservatively assumed to be 10% of the total iodine activity in the leaked fluid for the ECCS leakage entering the Reactor Auxiliary Building. For the ECCS leakage back to the RWT, the sump and RWT pH history and temperature are used to evaluate the amount of iodine that enters the RWT air space.
22. Regulatory Position 5.6 - For ECCS leakage into the Reactor Auxiliary Building, the form of the released iodine is 97% elemental and 3% organic. An ECCS area ventilation system filter efficiency of 95% is assumed for both elemental and organic iodine. The ECCS area ventilation system meets the requirements of RG 1.52 and Generic Letter 99-02. There is no credit for hold-up or dilution in the Reactor Auxiliary Building. The temperature and pH history of the sump and RWT are considered in determining the radioiodine available for release and the chemical form. Credit is taken for dilution of the activity in the RWT.
23. Regulatory Position 6 - This position relates to MSIV leakage in BWRs, which is not applicable to St. Lucie Unit 1.
24. Regulatory Position 7 - Containment purge is not considered as a means of combustible gas or pressure control in this analysis; however, the effect of containment purge before isolation is considered.

15.4.1.5.3 Methodology

Input assumptions used in the dose consequence analysis of a LOCA are provided in Table 15.4.1-6. For the purposes of the LOCA analyses, a major LOCA is defined as a rupture of the RCS piping, including the double-ended rupture of the largest piping in the RCS, or of any line connected to that system up to the first closed valve. Should a major break occur, depressurization of the RCS results in a pressure decrease in the pressurizer. A reactor trip signal occurs when the pressurizer low-pressure trip setpoint is reached. A SIS signal is actuated when the appropriate setpoint (high containment pressure) is reached. The following measures will limit the consequences of the accident in two ways:

1. Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat, and
2. Injection of borated water provides heat transfer from the core and prevents excessive cladding temperatures.

Release Inputs

The core inventory of the radionuclide groups utilized for this event is based on RG 1.183, Regulatory Position 3.1, at 100.3% of core thermal power and is provided as Table 15.4.1-1e. The source term represents end of cycle conditions assuming enveloping initial fuel enrichment and an average core burnup of 49,000 MWD/MTU.

The initial leakage rate from containment is 0.5% of the containment air per day. Per RG 1.183, Regulatory Position 3.7, the primary containment leakage rate is reduced by 50% at 24 hours into the LOCA to 0.25% /day based on the post-LOCA primary containment pressure history. The majority of the leakage is released through the plant stack via the SBVS. 9.6% if the total leakage is assumed to bypass the SBVS filters and is modeled as a release from a feedwater line penetration.

The ESF leakage to the Reactor Auxiliary Building is assumed to be 4750 cc/hr based upon two times the current licensing basis value of 2375 cc/hr. The leakage is conservatively assumed to start at 24 minutes into the event and continue throughout the 30-day period. This portion of the analysis assumes that 10% of the total iodine is released from the leaked liquid. The form of the released iodine is 97% elemental and 3% organic. Dilution and holdup of the ECCS leakage in the Reactor Auxiliary Building are not credited.

The ECCS backleakage to the RWT is initially assumed to be 2 gpm based upon doubling the current bounding value of 1 gpm. This leakage is assumed to start at 24 minutes into the event when recirculation starts and continue throughout the 30-day period. Based on sump pH history, the iodine in the sump solution is assumed to all be nonvolatile. However, when introduced into the acidic solution of the RWT inventory, there is a potential for the particulate iodine to convert into the elemental form. The fraction of the total iodine in the RWT which becomes elemental is both a function of the RWT pH and the total iodine concentration. The amount of elemental iodine in the RWT fluid which then enters the RWT air space is a function of the temperature-dependent iodine partition coefficient.

The time-dependent concentration of the total iodine in the RWT (including stable iodine) was determined from the tank liquid volume and leak rate. This iodine concentration ranged from a minimum value of 0 at the beginning of the event to a maximum value of 4.07E-05 gm-atom/liter at 30 days. Based upon the backleakage of sump water, the RWT pH slowly increases from an initial value of 4.5 to a maximum pH of 4.968 at 30 days. Using the time-dependent RWT pH and the total iodine concentration in the RWT liquid space, the amount of iodine converted to the elemental form was determined using guidance provided in NUREG/CR-5950 (Reference 105). This RWT elemental iodine fraction ranged from 0 at the beginning of the event to a maximum of 0.173.

The elemental iodine in the liquid region of the RWT is assumed to become volatile and to partition between the liquid and vapor space in the RWT based upon the partition coefficient for elemental iodine as presented in NUREG/CR-5950. A GOTHIC model was used to determine the RWT temperature as a function of time which was then used to calculate the partition coefficient. The RWT is a vented tank; therefore, there will be no pressure transient in the air region that would affect the partition coefficient. Since no boiling occurs in the RWT, the release of the activity from the vapor space within the RWT is calculated based upon the displacement of air by the incoming leakage. The elemental iodine flow rate from the RWT is equal to the air flow rate times the elemental iodine concentration in the RWT vapor space.

For the organic iodine flow, the same approach was used with an organic iodine fraction of 0.0015 from RG 1.183 in combination with a partition coefficient of 1.0. The particulate portion of the leakage is assumed to be retained in the liquid phase of the RWT. Therefore, the total iodine flow is the sum of the elemental and organic iodine flow rates.

The time dependent iodine release rate to the RWT vapor space presented in Table 15.4.1-8a is then applied to the entire iodine inventory (particulate, elemental and organic) in the containment sump. The iodine released via the RWT air vent to the environment Table 15.4.1-8b was effectively set to 100% elemental (the control room filters have the same efficiency for all forms of iodine).

Containment purge is also assumed coincident with the beginning of the LOCA. The Hydrogen Purge system is manually isolated within 285 seconds of the beginning of the event. The initial RCS activity (at an assumed 1.0 microcuries per gram DE I-131 and 518.9 μ Ci/gm DE XE-133 microcuries per gram gross activity) and fuel/gap release activity is modeled for 285 seconds at 500 cfm until isolation occurs.

Transport Inputs

During the LOCA event, the initial containment purge is released through the plant stack with no filtration. Leakage into the secondary containment is assumed to be released directly to the environment as a ground level release prior to drawdown of the secondary containment at 310 seconds. Activity subsequently collected by the SBVS is assumed to be a filtered release from the plant stack with a filter efficiency of 99% for particulates and 95% for both elemental and organic iodine. The activity that bypasses the SBVS is released unfiltered to the environment via a ground level release from containment. ECCS leakage into the Auxiliary Building is modeled as a release via the Reactor Auxiliary Building. For this release path, the ECCS area ventilation system is credited with a particulate removal efficiency of 99% and elemental and organic iodine efficiencies of 95%. The activity from the RWT is modeled as an unfiltered ground level release from the RWT.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 460 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a CIAS as a result of a high containment pressure signal. A 50-second delay is applied to account for the time to reach the signal, the diesel generator start time, damper actuation time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 460 cfm of unfiltered inleakage, and 1760 cfm of filtered recirculation flow.
- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of up to 504 cfm of filtered makeup flow, 460 cfm of unfiltered inleakage, and 1256 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% elemental iodine, and 95% organic iodine.

LOCA Removal Inputs

Reduction of the airborne radioactivity in the containment by natural deposition is credited. The natural deposition removal coefficient for elemental iodine is calculated per SRP 6.5.2 as 2.89 hr⁻¹. A natural deposition removal coefficient of 0.1 hr⁻¹ is assumed for all aerosols in the unsprayed region and in the sprayed region after spray flow is secured at 8 hours. No removal of organic iodine by natural deposition is assumed.

Containment spray provides coverage to 86% of the containment. Therefore, the St. Lucie Unit 1

containment building atmosphere is not considered to be a single, well-mixed volume. A mixing rate of two turnovers of the unsprayed region per hour is assumed.

The elemental spray coefficient is limited to 20 hr⁻¹ per SRP 6.5.2. This coefficient is reduced to 0 when an elemental decontamination factor (DF) of 200 is reached. Based upon the elemental iodine removal rate of 20 hr⁻¹, the DF of 200 is conservatively computed to occur at 2.331 hours. The particulate iodine removal rate is reduced by a factor of 10 when a DF of 50 is reached. Based upon the calculated iodine aerosol removal rate of 6.07 hr⁻¹, the DF of 50 is conservatively computed to occur at 2.334 hours.

15.4.1.5.4 Radiological Consequences

The Control Room atmospheric dispersion factors (χ/Q s) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake.

When the Control Room Ventilation System is in normal mode, the most limiting χ/Q corresponds to the worst air intake to the control room. When the ventilation system is isolated at 50 seconds, the limiting χ/Q corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. Development of control room atmospheric dispersion factors is discussed in Appendix 2J. The χ/Q s for the LOCA releases are summarized in Table 15.4.1-10.

For the EAB dose analysis, the χ/Q factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period. The LPZ dose is determined using the χ/Q factors for the appropriate time intervals. These χ/Q factors are provided in Appendix 2I.

The radiological consequences of the design basis LOCA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. In addition, the MicroShield code, Version 5.05, Grove Engineering, is used to develop direct shine doses to the Control Room. MicroShield is a point kernel integration code used for general-purpose gamma shielding analysis.

The post accident doses are the result of five distinct activity releases:

- Containment Purge at event initiation.
- Containment leakage via the SBVS.
- Containment leakage which bypasses the SBVS.
- ESF system leakage into the Reactor Auxiliary Building.
- ESF system leakage into the RWT.

The dose to the Control Room occupants includes terms for:

1. Contamination of the Control Room atmosphere by intake and infiltration of radioactive material from the containment and ESF.
2. External radioactive plume shine contribution from the containment and ESF leakage releases. This term takes credit for Control Room structural shielding.

3. A direct shine dose contribution from the Containment's contained accident activity. This term takes credit for both Containment and Control Room structural shielding.
4. A direct shine dose contribution from the activity collected on the Control Room ventilation filters.

As shown in Table 15.4.1-11, the sum of the results of all dose contributions for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

The calculation of record for this analysis is in Section 15.5, Reference (8).

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15.4.1.6 Hydrogen Accumulation in Containment

A hydrogen control system consisting of a hydrogen purge system and a hydrogen sampling system, is included in the plant design to prevent excessive hydrogen buildup following a LOCA.

The results of an analysis of hydrogen accumulation in the containment following a major LOCA, show that hydrogen can be controlled by recombination or purging. These systems described in Section 6.2.5, will be used to maintain the maximum hydrogen volumetric concentration at or below 4 percent. The lower flammability limit for hydrogen in air saturated with water vapor at room temperature and atmospheric pressure is 4.1 volume percent (References 37, 38, and 39). For these conditions, a concentration of approximately 18- 19 percent is required for detonation.

These systems allow considerable operational flexibility. The specific mode of operation would be determined by the actual hydrogen generation rate, the hydrogen concentration in the containment atmosphere, the amount of airborne activity in the containment, and the prevailing meteorological conditions.

15.4.1.7 Effect of Replacement Steam Generators

The replacement steam generators (RSGs) have no effect on the blowdown phase of a LBLOCA because small variations in steam generator parameters have an insignificant effect on the blowdown rate, which is driven by critical flow through the break. The RSGs have no effect on the adiabatic heatup phase of the transient because the rate of clad temperature heatup is a function of core decay heat and the duration of the phase is a function of ECCS flow rates. The RSGs, with 25 percent tube plugging (18 percent plus 7 percent plugging asymmetry), have a lower flow resistance than the model of the original steam generator, with 32 percent tube plugging, that was used in the LBLOCA analyses. Consequently, the PCTs calculated using the current evaluation model remain bounding with the RSGs. Because cladding temperatures calculated by the current evaluation model are bounding, so are the local and whole-core cladding oxidations. Likewise, the conclusions in the UFSAR regarding maintenance of core cooling remain valid.

TABLE 15.4.1-1

SAMPLED LBLOCA PARAMETERS

Phenomenological	
	Time in cycle (peaking factors, axial shape, rod properties and burnup)
	Break Type (guillotine versus split)
	Critical flow discharge coefficients (break)
	Decay heat ¹
	Critical flow discharge coefficients (surge line)
	Initial upper head temperature
	Film boiling heat transfer
	Dispersed film boiling heat transfer
	Critical heat flux
	T _{min} (intersection of film and transition boiling)
	Initial stored energy
	Downcomer hot wall effects
	Steam generator interfacial drag
	Condenser interphase heat transfer
	Metal-water reaction
Plant ²	
	Offsite power availability ³
	Break Size
	Pressurizer Pressure
	Pressurizer liquid level
	SIT pressure
	SIT liquid level
	SIT temperature (based on containment temperature)
	Containment temperature
	Containment volume
	Initial RCS flow rate
	Initial operating RCS temperature
	Diesel start (for loss of offsite power only)

¹ Not sampled in analysis, multiplier set to 1.0.

² Uncertainties for plant parameters are based on plant-specific values with the exception of "Offsite power availability," which is binary result that is specified by the analysis methodology.

³ This is no longer a sampled parameter. One set of 59 cases is run with LOOP and one set of 59 cases is run with no-LOOP.

TABLE 15.4.1-1a

PLANT OPERATING RANGE SUPPORTED BY THE LOCA ANALYSIS

	EVENT	OPERATING RANGE
1.0	PLANT PHYSICAL DESCRIPTION	
	<u>1.1 Fuel</u>	
	a) Cladding outside diameter b) Cladding inside diameter c) Cladding thickness d) Pellet outside diameter e) Pellet density f) Active fuel length g) Gd ₂ O ₃ concentrations	0.440 in. 0.384 in. 0.028 in. 0.377 in. 95.35 percent of theoretical 136.7 in. 2, 4, 6, 8 w/o
	<u>1.2 RCS</u>	
	a) Flow resistance b) Pressurizer location c) Hot assembly location d) Hot assembly type e) SG tube plugging	Analysis Analysis assumes location giving most limiting PCT (broken loop) Anywhere in core 14X14 Framatome (formerly AREVA) HTP fuel 10 percent
2.0	PLANT INITIAL OPERATING CONDITIONS	
	<u>2.1 Reactor Power</u>	
	a) Nominal reactor power b) LHGR c) F _Q d) Fr	3029.1 MWt ¹ 15.0 kW/ft 2.08175 - 2.161 1.810 ²
	<u>2.2 Fluid Conditions</u>	
	a) Loop flow b) RCS Cold Leg temperature c) Pressurizer pressure d) Pressurizer level e) SIT pressure f) SIT liquid volume g) SIT temperature h) SIT resistance fL/D i) Minimum ECCS boron	140.8 Mlbm/hr ≤ M ≤ 164.6 Mlbm/hr 548.0°F ≤ T ≤ 554.0°F 2185 psia ≤ P ≤ 2315 psia 62.6 percent ≤ L ≤ 68.6 percent 214.7 psia ≤ P ≤ 294.7 psia 1090 ft ³ ≤ V ≤ 1170 ft ³ 80.5°F ≤ T ≤ 124.5°F (It couples with containment temperature) As-built piping configuration ≥1900 ppm

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1. Includes 0.3% uncertainties
2. The radial power peaking for the hot rod is including 6% measurement uncertainty and 3.5% allowance for control rod insertion affect.

TABLE 15.4.1-1a

PLANT OPERATING RANGE SUPPORTED BY THE LOCA ANALYSIS

(Continued)

	EVENT	OPERATING RANGE
3.0	ACCIDENT BOUNDARY CONDITIONS	
	a) Break location	Cold leg pump discharge piping
	b) Break type	Double-ended guillotine or split
	c) Break size (each side, relative to cold leg pipe area)	0.2997 ≤ A ≤ 1.0 full pipe area (split) 0.2997 ≤ A ≤ 1.0 full pipe area (guillotine)
	d) Worst single-failure	Loss of one emergency diesel generator
	e) Offsite power	On or Off
	f) LSPI flow	Minimum flow
	g) HPSI flow	Minimum flow
	h) ECCS pumped injection temperature	120°F
	i) HPSI pump delay	19.5 sec (w/offsite power) 30.0 sec w/o offsite power
	j) LPSI pump delay	19.5 sec (w/offsite power) 30.0 sec w/o offsite power
	k) Containment pressure	14.7 psia, nominal value
	l) Containment temperature	80.5°F ≤ T ≤ 124.5°F
	m) Containment sprays delay	0 sec
	n) Containment volume	2.46078E+06 ft ³ ≤ V ≤ 2.63655E+06 ft ³

TABLE 15.4.1-1b

STATISTICAL DISTRIBUTIONS USED FOR PROCESS PARAMETERS

PARAMETER	OPERATIONAL UNCERTAINTY DISTRIBUTION	PARAMETER RANGE
Pressurizer Pressure (psia)	Uniform	2185 - 2315
Pressurizer Liquid Level (percent)	Uniform	62.6 – 68.6
SIT Liquid Volume (ft ³)	Uniform	1090.0 – 1170.0
SIT Pressure (psia)	Uniform	214.7 – 294.7
Containment Temperature (°F)	Uniform	80.5- 124.5
Containment Volume (ft ³)	Uniform	2.46078E+6 – 2.63655E+6
Initial RCS Flow Rate (Mlbm/hr)	Uniform	140.8 – 164.6
Initial RCS Operating Temperature (Tcold) (°F)	Uniform	548.0 – 554.0
Offsite Power Availability ¹	Binary	0,1

¹ This is no longer a sampled parameter. One set of 59 cases is run with LOOP and one set of 59 cases is run with No-LOOP.

TABLE 15.4.1-1c

SUMMARY OF MAJOR PARAMETERS FOR THE LIMITING PCT CASE

	Fresh UO ₂ Fuel	Once-Burned UO ₂ Fuel
Cycle Burnup (EFPH)	343.73	343.68
Core Power (MWt)	3029.1	3029.1
Maximum Hot Rod LHGR (kW/ft)	15.157	14.702
Radial Peak (F _r)	1.810	1.756
Axial Offset	0.0620	0.0620
Break Type	Guillotine	Guillotine
Break Size (ft ² /side)	3.2791	3.2791
Offsite Power Availability	Not available	Not available
Decay Heat Multiplier	1.0	1.0

TABLE 15.4.1-1d

CALCULATED EVENT TIMES FOR THE LIMITING PCT CASE

EVENT	TIME(S)
Break Opened	0.0
RCP Trip	0.0
SIAS Issued	1.1
Start of Broken Loop SIT Injection	16.7
Start of Intact Loop SIT Injection (Loops 2,3 and 4 respectively)	19.3, 19.4 and 19.4
Broken Loop LPSI Delivery Began	31.1
Intact Loop LPSI Delivery Began (Loops 2, 3, and 4 respectively)	N/A, N/A and 31.1
Broken Loop HPSI Delivery Began	31.1
Intact Loop HPSI Delivery Began (Loops 2, 3, and 4 respectively)	31.1, 31.1 and N/A
Beginning of Core Recovery (Beginning of Reflood)	28.8
Broken Loop SIT Emptied	63.1
Intact Loop SITs Emptied (Loops 2, 3, and 4 respectively)	60.7, 63.4 and 66.3
PCT Occurred	51.5
Transient Calculation Terminated	605.0

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TABLE 15.4.1-1e

LOCA CONTAINMENT LEAKAGE SOURCE TERM

Nuclide	Core/Fuel Source (Curies)	Nuclide	Core/Fuel Source (Curies)
Kr-85	1.238E+06	Pu-239	3.828E+04
Kr-85m	1.983E+07	Pu-240	7.207E+04
Kr-87	3.767E+07	Pu-241	1.785E+07
Kr-88	5.295E+07	Am-241	2.014E+04
Rb-86	2.817E+05	Cm-242	8.940E+06
Sr-89	7.261E+07	Cm-244	3.272E+06
Sr-90	9.934E+06	I-130	6.937E+06
Sr-91	9.016E+07	Kr-83m	9.565E+06
Sr-92	9.856E+07	Xe-138	1.320E+08
Y-90	1.036E+07	Xe-131m	9.824E+05
Y-91	9.485E+07	Xe-133m	5.358E+06
Y-92	9.904E+07	Xe-135m	3.513E+07
Y-93	1.158E+08	Cs-138	1.470E+08
Zr-95	1.337E+08	Cs-134m	7.473E+06
Zr-97	1.330E+08	Rb-88	5.392E+07
Nb-95	1.352E+08	Rb-89	6.883E+07
Mo-99	1.581E+08	Sb-124	3.526E+05
Tc-99m	1.384E+08	Sb-125	2.324E+06
Ru-103	1.578E+08	Sb-126	1.787E+05
Ru-105	1.277E+08	Te-131	7.697E+07
Ru-106	9.086E+07	Te-133	9.845E+07
Rh-105	1.150E+08	Te-134	1.312E+08
Sb-127	1.163E+07	Te-125m	5.143E+05
Sb-129	3.155E+07	Te-133m	5.818E+07
Te-127	1.157E+07	Ba-141	1.304E+08
Te-127m	1.578E+06	Ba-137m	1.312E+07
Te-129	3.105E+07	Pd-109	5.544E+07
Te-129m	4.607E+06	Rh-106	9.960E+07
Te-131m	1.330E+07	Rh-103m	1.422E+08
Te-132	1.213E+08	Tc-101	1.470E+08
I-131	8.752E+07	Eu-154	2.086E+06
I-132	1.240E+08	Eu-155	1.446E+06
I-133	1.650E+08	Ei-156	4.763E+07
I-134	1.787E+08	La-143	1.198E+08
I-135	1.555E+08	Nb-97	1.342E+08

TABLE 15.4.1-1e
LOCA CONTAINMENT LEAKAGE SOURCE TERM (CONT.)

Nuclide	Core/Fuel Source (Curies)	Nuclide	Core/Fuel Source (Curies)
Xe-133	1.657E+08	Nb-95m	9.559E+05
Xe-135	4.394E+07	Pm-147	1.212E+07
Cs-134	3.335E+07	Pm-148	2.472E+07
Cs-136	8.190E+06	Pm-149	5.555E+07
Cs-137	1.384E+07	Pm-151	2.031E+07
Ba-139	1.439E+08	Pm-148m	2.971E+06
Ba-140	1.386E+08	Pr-144	1.129E+08
La-140	1.448E+08	Pr-144m	1.347E+06
La-141	1.311E+08	Sm-153	6.783E+07
La-142	1.263E+08	Y-94	1.175E+08
Ce-141	1.333E+08	Y-95	1.272E+08
Ce-143	1.207E+08	Y-91m	5.234E+07
Ce-144	1.121E+08	Br-82	7.734E+05
Pr-143	1.200E+08	Br-83	9.531E+06
Nd-147	5.290E+07	Br-84	1.632E+07
Np-239	2.435E+09	Am-242	1.235E+07
Pu-238	6.206E+05	Np-238	6.601E+07
		Pu-243	1.146E+08

TABLE 15.4.1-1f
CONTAINMENT HEAT SINK DATA

DESCRIPTION	SLAB MATERIAL	MATERIAL THICK. (ft)	AREA (ft ²)
Containment Shell	C Steel	0.1171	86700
Floor Slab	Concrete	21.0	12682
Misc Concrete	Concrete	1.5	87751
Galvanized Steel	Galvanizing C Steel	0.0005833 0.01417	130000
Carbon Steel	C Steel	0.03125	25000
Stainless Steel	S Steel	0.0375	22300
Misc Steel	C Steel	0.0625	40000
Misc Steel	C Steel	0.02083	41700
Misc Steel	C Steel	0.17708	7000
Imbedded Steel	C Steel Concrete	0.0708 7.0	18000
Sump (GSI-191)	C Steel	0.02895	7414

TABLE 15.4.1-g

CONTAINMENT INITIAL AND BOUNDARY CONDITIONS

PARAMETER	PARAMETER VALUE
Containment free volume range, ft ³	2,460,780 - 2,636,550
Initial relative humidity	100.0%
Initial compartment pressure, psia	14.7, nominal value
Initial compartment temperature, °F	80.5 ≤ T ≤ 124.5
Containment spray time of delivery, sec	0.0
Containment spray flow rate, gpm	9,000.0
Containment spray temperature, °F	36.0

TABLE 15.4.1-h

SUMMARY OF RESULTS FOR THE LIMITING PCT CASE

Case #23 (Offsite Power Not Available)	14 X 14 Framatome (formerly AREVA)	
	Fresh Fuel UO ₂ Rod	Once-Burned Fuel UO ₂ Rod
PCT		
Temperature	1788°F	1774°F
Time	51.5s	51.486s
Elevation	7.859 ft	7.8587 ft
Metal-Water Reaction		
Pre-transient Oxidation %	0.1992	0.666
Transient Oxidation Maximum %	1.6551	1.5602
Total Oxidation Maximum %	1.854	2.226
Total Whole-Core Oxidation %	0.0392	N/A

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TABLE 15.4.1-2

MAXIMUM POTENTIAL RECIRCULATION LOOP LEAKAGE
(OUTSIDE CONTAINMENT)

<u>Item</u>	<u>Number of Units</u>	<u>Type of Leakage Control</u>	<u>Unit Leakage Rate Used</u>	<u>Leakage to ECCS Pump Room (cc/hr)</u>	<u>Leakage to Equipment Drain Tank (cc/hr)</u>
Containment Spray Pumps	2	Mechanical Seals with leakoff	0.5 cc/minute	60	0
Low Pressure Safety Injection Pumps	2	Mechanical Seals with leakoff	0.5 cc/minute	60	0
High Pressure Safety Injection Pumps	3	Mechanical Seals with leakoff	0.5 cc/minute	90	0
Pump Flanges	14	Gasket adjusted to zero leakage following any test	0.5 cc/minute	420	0
Valve Flanges	63	Gasket adjusted to zero leakage following any test	0.2 cc/minute	756	0
Blind and Orifice Flanges	30	Gasket adjusted to zero leakage following any test	0.5 cc/minute	900	0
Large Valves 2 1/2 inches	63	Double packing with leakoff	1 cc/hour	63	0
Small Valves	26	Packed stems	1.0 cc/hour	<u>26</u>	<u>0</u>
			Totals	2,375	0

15.4.1-20

Amendment No. 26 (11/13)

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TABLE 15.4.1-5
CLASS 3 - DESIGN BASIS ACCIDENT OFF-SITE DOSES
(HISTORICAL)

Accident Body	Time Period	Distance	Dose Using Applicant's Model (Rem)		Dose using AEC Safety Guide 4 Model (Rem)	
			Thyroid	Whole Body	Thyroid	Whole
*LOCA (Base Case)	0-2 hr	5100 ft	.236	1.52x10 ⁻³	66.7	2.05
	0-31 day	5 miles	1.56	4.93x10 ⁻³	42.9	0.622
*LOCA	0-2 hr	5100 ft	.326	2.61x10 ⁻³	63.3	2.03
	0-31 day	5 miles	2.15	8.09x10 ⁻³	40.7	0.618
*Bypass Leakage	0-2 hr	5100 ft	----	----	5.2	3.5x10 ⁻²
	0-31 day	5 miles	----	----	4.2	1.5x10 ⁻²
Hydrogen Purge (Base Case)	Long Term	5100 ft	1.06x10 ⁻³	2.31x10 ⁻⁴	0.55	6.0x10 ⁻⁴
	Long Term	5 miles	8.14x10 ⁻⁵	1.78x10 ⁻⁵	0.050	4.69x10 ⁻⁵
Hydrogen Purge	Long Term	5100 ft	1.02x10 ⁻³	9.04x10 ⁻⁴	0.53	2.10x10 ⁻³
	Long Term	5 miles	7.85 10 ⁻⁵	6.96x10 ⁻⁵	0.03	1.64x10 ⁻⁴
ESF Component Leakage (Base Case)	0-2 hr	5100 ft	----	----	0.0796	----
	0-31 day	5 miles	----	----	0.0714	----
ESF Component Leakage (Extended Burnup)	0-2 hr	5100 ft	----	----	0.0793	----
	0-31 day	5 miles	----	----	0.0711	----
CEA Ejection:						
Containment Release	0-2 hr	EAB	----	----	0.066	6.9x10 ⁻⁵
	0-31 day	LPZ	----	----	0.639	6.9x10 ⁻⁴
Secondary Release	0-2 hr	EAB	----	----	0.525	4.35x10 ⁻⁴
	0-31 day	LPZ	----	----	1.131	9.3x10 ⁻⁴
Steam Line Break ⁽¹⁾	0-2 hr	----	----	----	----	----
Steam Generator Tube Break	0-2 hr	5100 ft	0.863	0.217	----	----
Waste Gas ⁽²⁾	----	----	----	----	----	----
Fuel Handling Cont. (Ext. Burnup)	0-2 hr	5100 ft	59.1 rem	0.68 rem		
	0-2 hr	5 miles	27.9 rem	0.319 rem		
10 CFR 100 Limits	0-2 hr	5100 ft	300	25	300	25
10 CFR 100 Limits	0-31 day	5 miles	300	25	300	25

1. The doses for Steam Line Break are bounded by the LOCA doses.

2. The doses for Waste Gas Decay Tank Accident are bounded by the Fuel Handling Accident doses.

*Historical information. LOCA dose consequences revised using more conservative γ/Q values and a one (1) mile LPZ in support of Technical Specification Amendment #38. See discussion in UFSAR Section 15.4.1.5.

Table 15.4.1-6

Loss-of-Coolant Accident (LOCA) – Inputs and Assumptions

Input/Assumption	Value
Release Inputs:	
Core Power Level	3030 MW _{th} (3020 + 0.3%)
Core Average Fuel Burnup	49,000 MWD/MTU
Fuel Enrichment	1.5 – 5.0 w/o
Initial RCS Equilibrium Activity	1.0 μCi/gm DE I-131 and 518.9 μCi/gm DE Xe-133 gross activity (Table 15.4.1-9)
Core Fission Product Inventory	Table 15.4.1-1e
Containment Leakage Rate 0 to 24 hours after 24 hours	0.5% (by volume)/day 0.25% (by volume)/day
LOCA release phase timing and duration	Table 15.4.1-7
Core Inventory Release Fractions (gap release and early in-vessel damage phases)	RG 1.183, Sections 3.1 and 3.2
<u>ECCS Systems Leakage</u>	
Sump Volume (minimum)	67,394 ft ³
ECCS Leakage to RAB (2 times allowed value)	4750 cc/hr
Flashing Fraction	Calculated – 5.5% Used for dose determination – 10%
Chemical form of the iodine in the sump water	0% aerosol, 97% elemental, and 3.0% organic
Release ECCS Area Filtration Efficiency	Elemental – 95% Organic – 95% Particulate – 99% (100% of the particulates are retained in the ECCS fluid)

Table 15.4.1-6

Loss-of-Coolant Accident (LOCA) – Inputs and Assumptions (Cont'd)

Input/Assumption	Value
<u>RWT Back-leakage</u>	
Sump Volume (at time of recirculation)	67,394 ft ³
ECCS Leakage to RWT (2 times allowed value)	2 gpm
Flashing Fraction (elemental iodine assumed to be released into tank space based upon partition factor)	0 % based on temperature of fluid reaching RWT
Initial RWT Liquid Inventory (minimum)	44,147 gal
Release from RWT Vapor Space	Table 15.4.1-8
Containment Purge Release	500 cfm for 30 seconds
Removal Inputs:	
Containment Particulate/Aerosol Natural Deposition (only credited in unsprayed regions)	0.1/hour
Containment Elemental Iodine Natural/Wall Deposition	2.89/hour
Containment Spray Region Volume	2,155,160 ft ³
Containment Unsprayed Region Volume	350,840 ft ³
Flowrate between sprayed and unsprayed volumes	23,389 cfm
Spray Removal Rates: Elemental Iodine Time to reach DF of 200 Particulate Iodine Time to reach DF of 50	20/hour 2.331 hours 6.07/hour 2.334 hours
Spray Initiation Time Spray Termination Time	80 seconds 8 hours
Control Room Ventilation System Time of automatic control room isolation Time of manual control room unisolation	50 seconds 1.5 hrs
Secondary Containment Filter Efficiency	Particulate – 99% Elemental – 95% Organic – 95%
Secondary Containment Drawdown Time	310 seconds
Secondary Containment Bypass Fraction	9.6%
Containment Purge Filtration	0 %
Transport Inputs:	
Containment Release Secondary Containment release prior to drawdown	Nearest containment penetration to CR ventilation intake

Table 15.4.1-6

Loss-of-Coolant Accident (LOCA) – Inputs and Assumptions (Cont'd)

Input/Assumption	Value
Containment Release Secondary Containment release after drawdown	Plant stack
Containment Release Secondary Containment Bypass Leakage	Nearest containment penetration to CR ventilation intake
ECCS Leakage	ECCS exhaust louver
RWT Backleakage	RWT
Containment Purge	Plant Stack
Personnel Dose Conversion Inputs:	
Atmospheric Dispersion Factors Offsite Onsite	Appendix 2I Table 15.4.1-10
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 15.4.1-7

LOCA Release Phases *

Phase	Onset	Duration
Gap Release	30 seconds	0.5 hours
Early In-Vessel	0.5 hours	1.3 hours

* From RG 1.183, Table 4

Table 15.4.1-8

Adjusted Sump to RWT Leakage Flow Rate

Time (hours)	Flow Rate (cfm)
0.0	0.0
0.40	7.973E-07
10.0	8.637E-06
25.0	4.886E-05
75.0	1.545E-04
125.0	2.636E-04
200.0	3.895E-04
300.0	4.995E-04
450.0	5.563E-04
600.0	5.687E-04

Table 15.4.1-8a

RWT Leakage Flow Rate

Time (hours)	Flow Rate (cfm)
0.0	1.07
720	1.07

Table 15.4.1-9

Reactor Coolant Source Term

Nuclide	RCS Activity ($\mu\text{Ci/gm}$)	Nuclide	RCS Activity ($\mu\text{Ci/gm}$)
Co-58	4.394E-02	Ba-139	4.800E-04
Co-60	8.256E-02	Ba-140	7.124E-03
Kr-85	4.160E+01	La-140	3.299E-03
Kr-85m	1.268E+00	La-141	2.646E-04
Kr-87	7.574E-01	La-142	7.483E-05
Kr-88	2.247E+00	Ce-141	1.147E-03
Rb-86	3.403E-02	Ce-143	6.485E-04
Sr-89	5.691E-03	Ce-144	9.815E-04
Sr-90	5.459E-04	Pr-143	1.031E-03
Sr-91	1.526E-03	Nd-147	4.302E-04
Sr-92	6.105E-04	Kr-83m	3.826E-01
Y-90	7.799E-04	Xe-138	5.168E-01
Y-91	3.327E-02	Xe-131m	3.022E+00
Y-92	7.402E-04	Xe-133m	3.479E+00
Y-93	4.706E-04	Xe-135m	8.553E-01
Zr-95	1.159E-03	Cs-138	9.667E-01
Zr-97	5.170E-04	Cs-134m	1.179E-01
Nb-95	1.187E-03	Rb-88	2.326E+00
Mo-99	5.111E+00	Rc-89	1.022E-01
Tc-99m	3.885E+00	Sb-124	1.910E-03
Ru-103	1.359E-03	Sb-125	1.277E-02
Ru-105	1.934E-04	Sb-126	9.173E-04
Ru-106	7.973E-04	Te-131	1.741E-02
Ru-105	7.047E-04	Te-134	2.280E-02
Sb-127	5.187E-02	Te-125m	2.828E-03
SB-129	2.918E-02	Te-133m	1.326E-02
Te-127	5.496E-02	Ba-141	1.009E-04
Te-127m	8.677E-03	Rh-103m	1.350E-03
Te-129	4.558E-02	Nb-97	8.889E-05
Te-129m	2.486E-02	Nb-95m	8.343E-06
Te-131m	4.273E-02	Pm-147	1.069E-04
Te-132	5.233E-01	Pm-148	1.866E-04
I-131	8.425E-01	Pm-149	3.485E-04
I-132	1.689E-01	Pm-151	1.021E-04
I-133	8.713E-01	Pm-148m	2.556E-05
I-134	7.726E-02	Pr-144	9.816E-04
I-135	3.933E-01	Y-94	1.528E-05
Xe-133	2.381E+02	Y-91m	8.872E-04
Xe-135	9.235E+00	Br-82	6.096E-02

Table 15.4.1-9

Reactor Coolant Source Term (Cont'd)

Nuclide	RCS Activity ($\mu\text{Ci/gm}$)	Nuclide	RCS Activity ($\mu\text{Ci/gm}$)
Cs-134	6.972E+00	Br-83	1.214E-01
Cs-136	1.543E+00	Br-84	5.002E-02
Cs-137	2.899E+00		

Table 15.4.1-10

Control Room χ/Q for SBVS Bypass Leakage

Time (hours)	χ/Q (sec/m³)
0	7.29E-03
0.01389	3.17E-03
1.5	1.76E-03
2	1.41E-03
8	5.72E-04
24	4.29E-04
96	3.57E-04
720	3.57E-04

Control Room χ/Q for SBVS Leakage and Containment Purge

Time (hours)	χ/Q (sec/m³)
0	2.39E-03
0.01389	3.91E-03
1.5	6.93E-04
2	4.88E-04
8	2.19E-04
24	1.46E-04
96	1.28E-04
720	1.28E-04

Control Room χ/Q for RWT Leakage

Time (hours)	χ/Q (sec/m³)
0	4.80E-03
0.01389	5.03E-03
1.5	3.61E-03
2	2.87E-03
8	1.20E-03
24	9.07E-04
96	7.13E-04
720	7.13E-04

Table 15.4.1-10 (Cont'd)

Control Room χ/Q for RWT Leakage

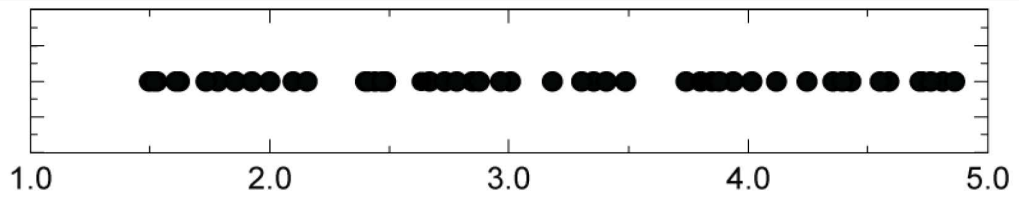
Time (hours)	χ/Q (sec/m³)
0	1.37E-03
0.01389	1.34E-03
1.5	1.12E-03
2	9.10E-04
8	3.84E-04
24	2.93E-04
96	2.37E-04
720	2.37E-04

Table 15.4.1-11

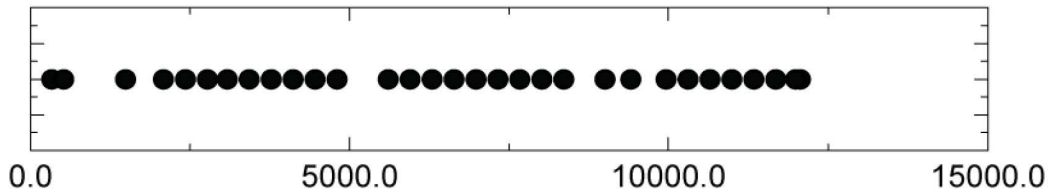
LOCA Dose Summary

Dose Contribution	TEDE Dose (rem)		
	EAB worst 2-hr	LPZ 30 days	CR 30 days
Containment Purge	3.8649E-04	3.7779E-04	5.7379E-03
Containment Leakage	1.1335E+00	2.4687E+00	4.3143E+00
ECCS Leakage	1.4956E-03	1.4751E-02	1.5237E-01
RWT Leakage	1.3722E-03	3.0891E-02	1.1503E-01
Shine Dose			0.20
Total	1.14	2.51	4.79
Acceptance Criteria	25	25	5

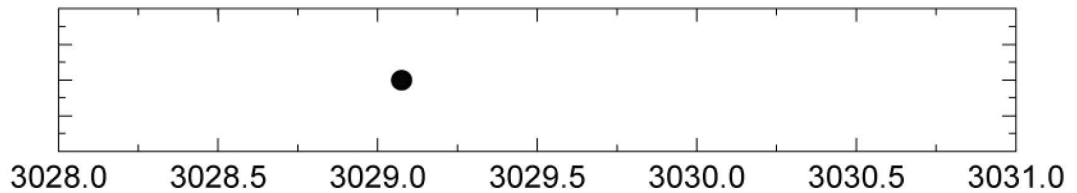
One-Sided
Break Area
(ft²/side)



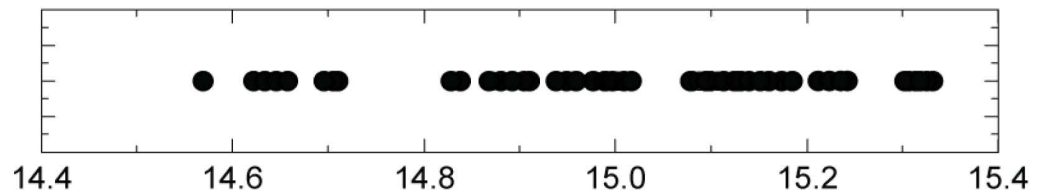
Burn
Time
(hours)



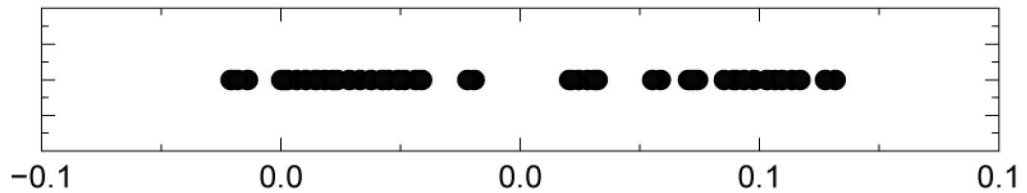
Core
Power
(MW)



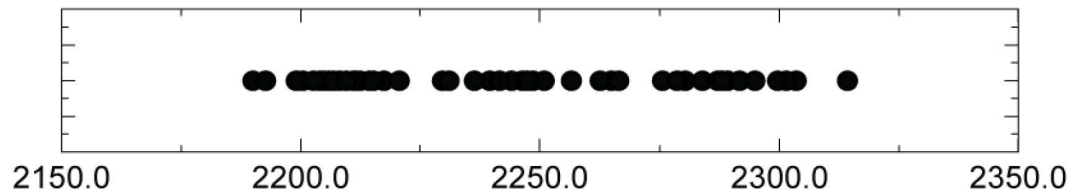
LHGR
(KW/ft)



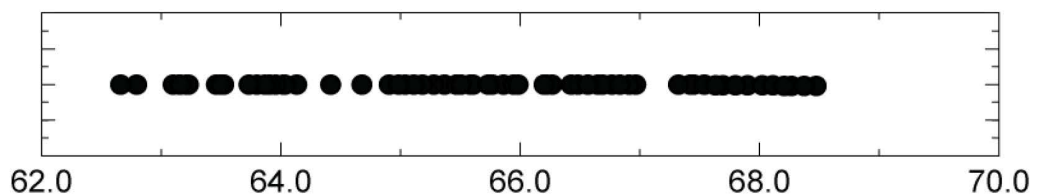
ASI



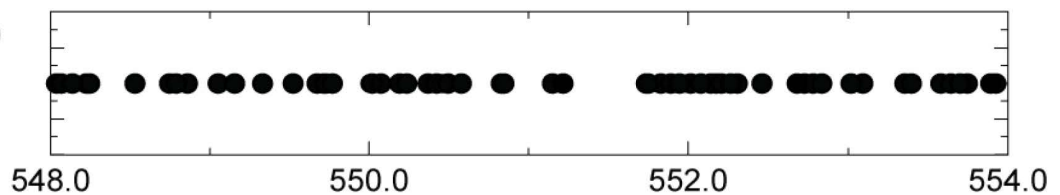
Pressurizer
Pressure
(psia)



Pressurizer
Liquid Level
(%)



RCS (Tcold)
Temperature
(°F)

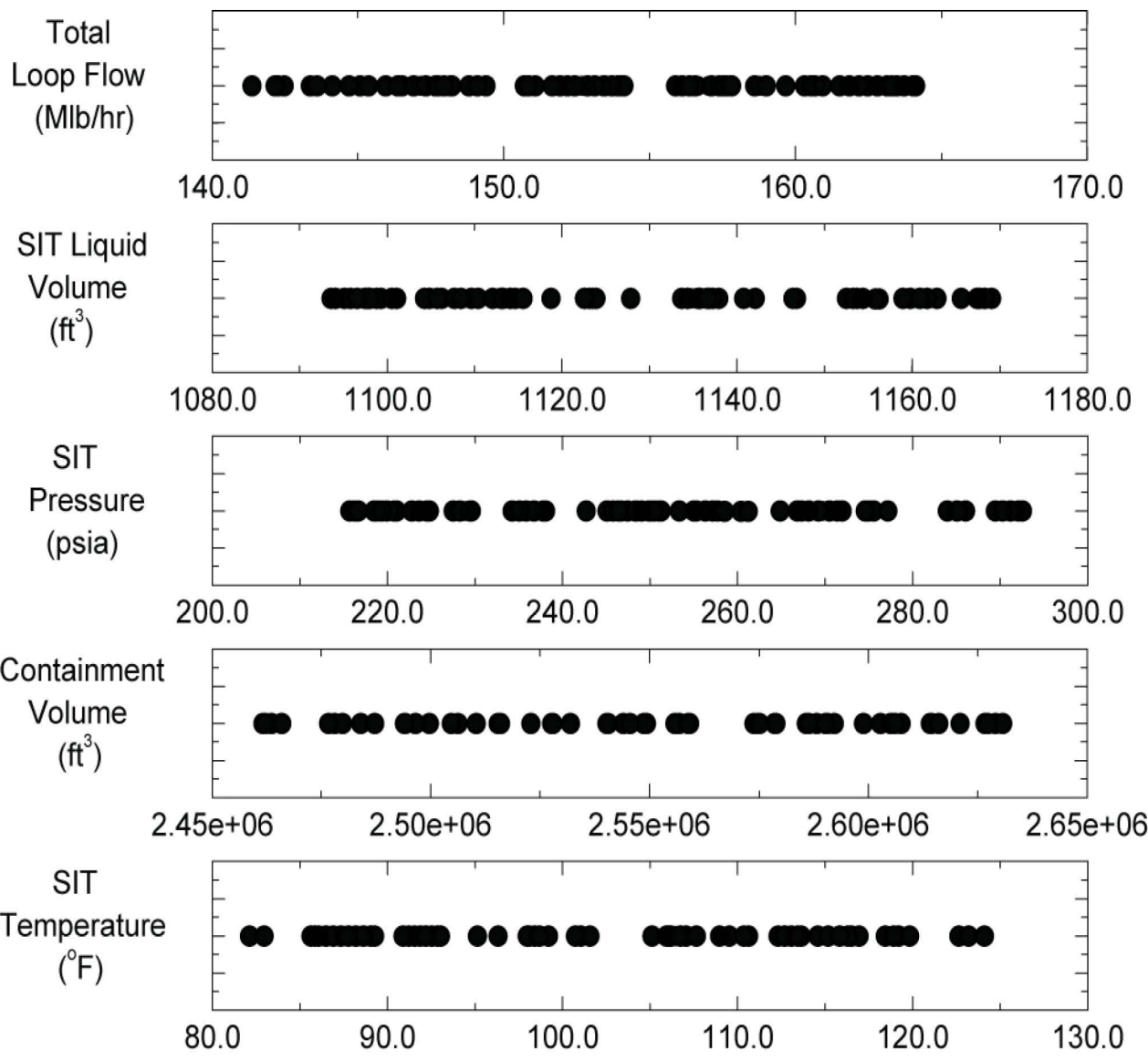


Amendment No. 27 (04/15)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

SCATTER PLOT OF OPERATIONAL
PARAMETERS

FIGURE 15.4.1-1

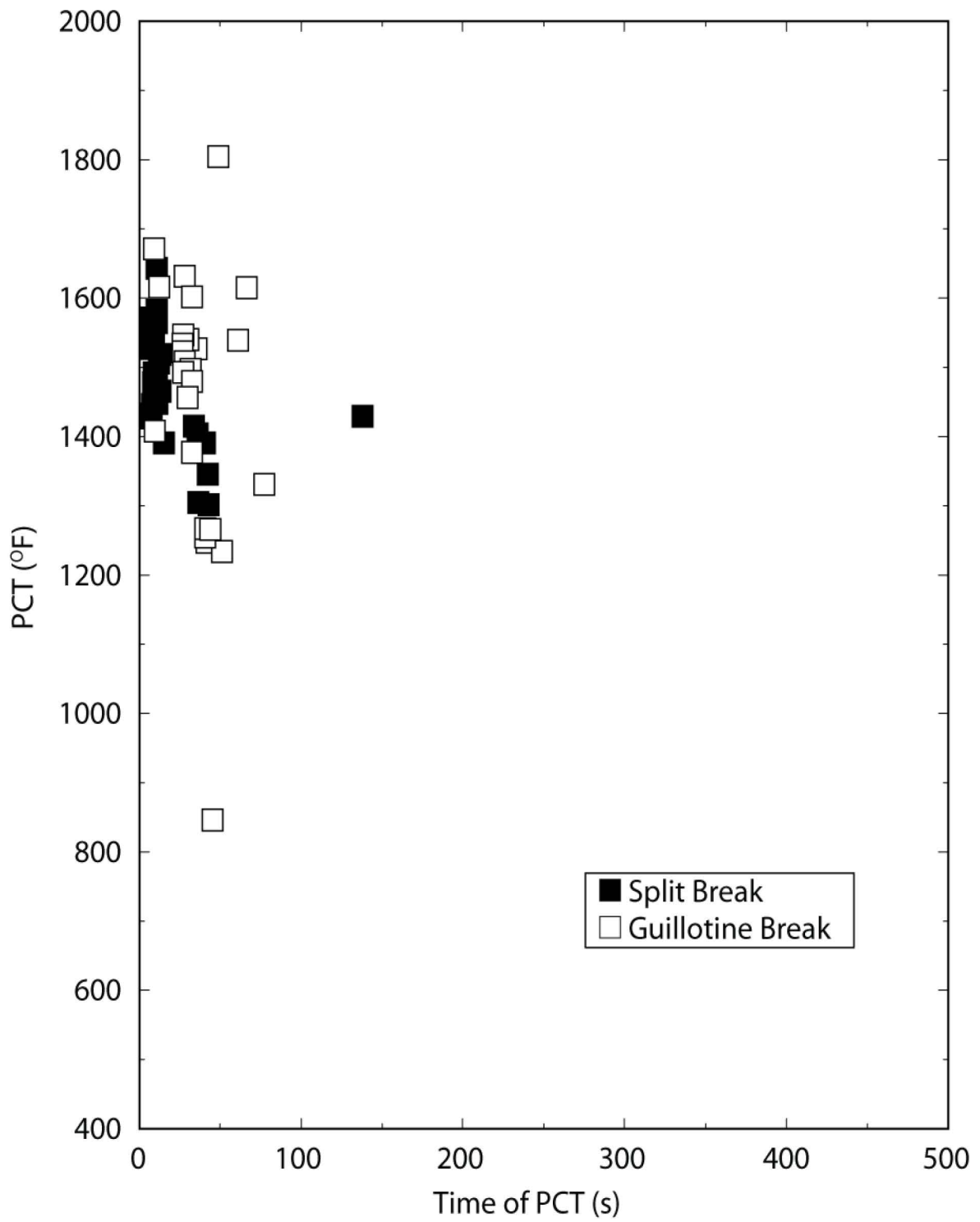


Amendment No. 27 (04/15)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

SCATTER PLOT OF OPERATIONAL
PARAMETERS (CONTINUED)

FIGURE 15.4.1-1a

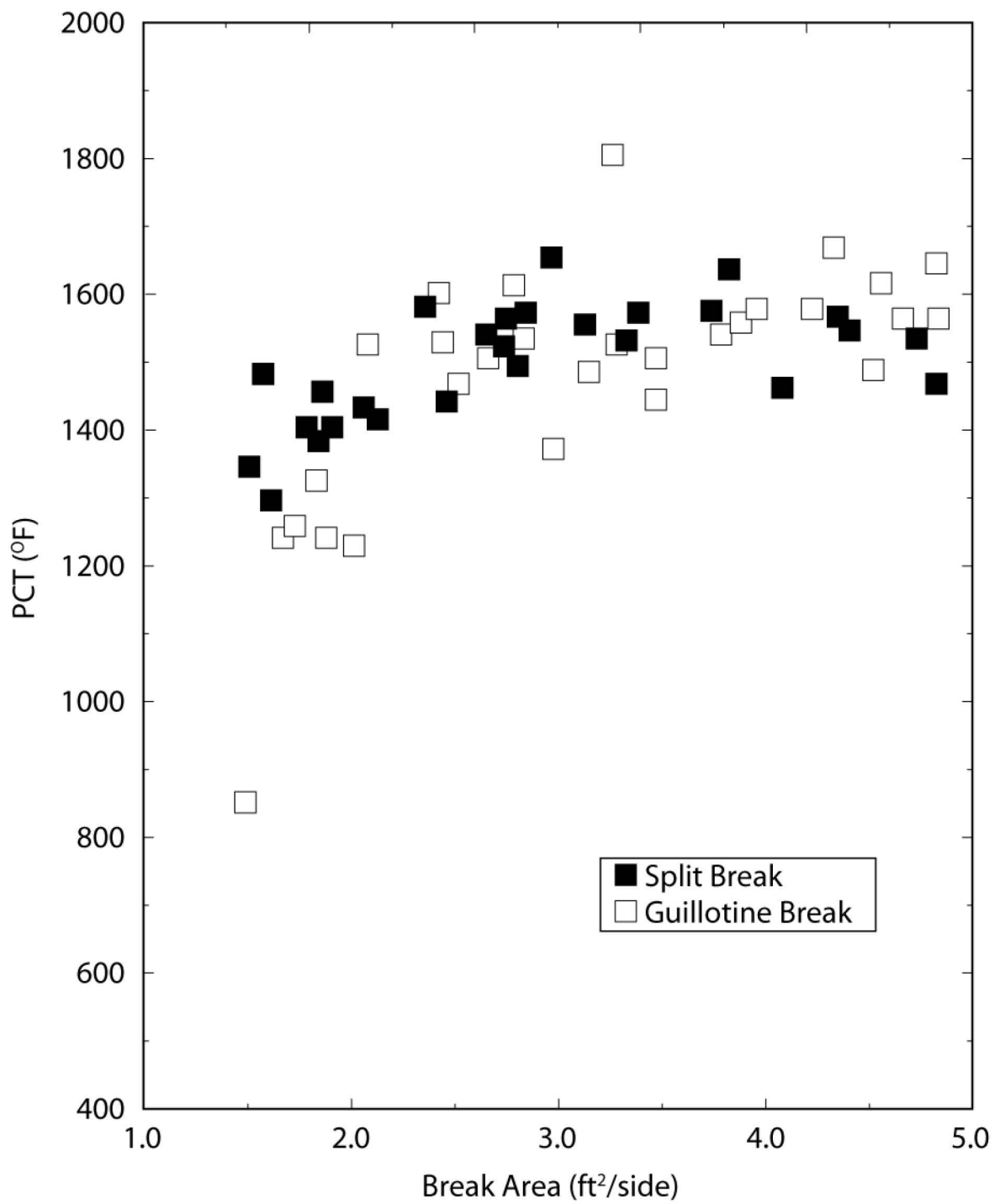


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**PCT VS PCT TIME
SCATTER PLOT FROM 59 CALCULATIONS**

FIGURE 15.4.1-2

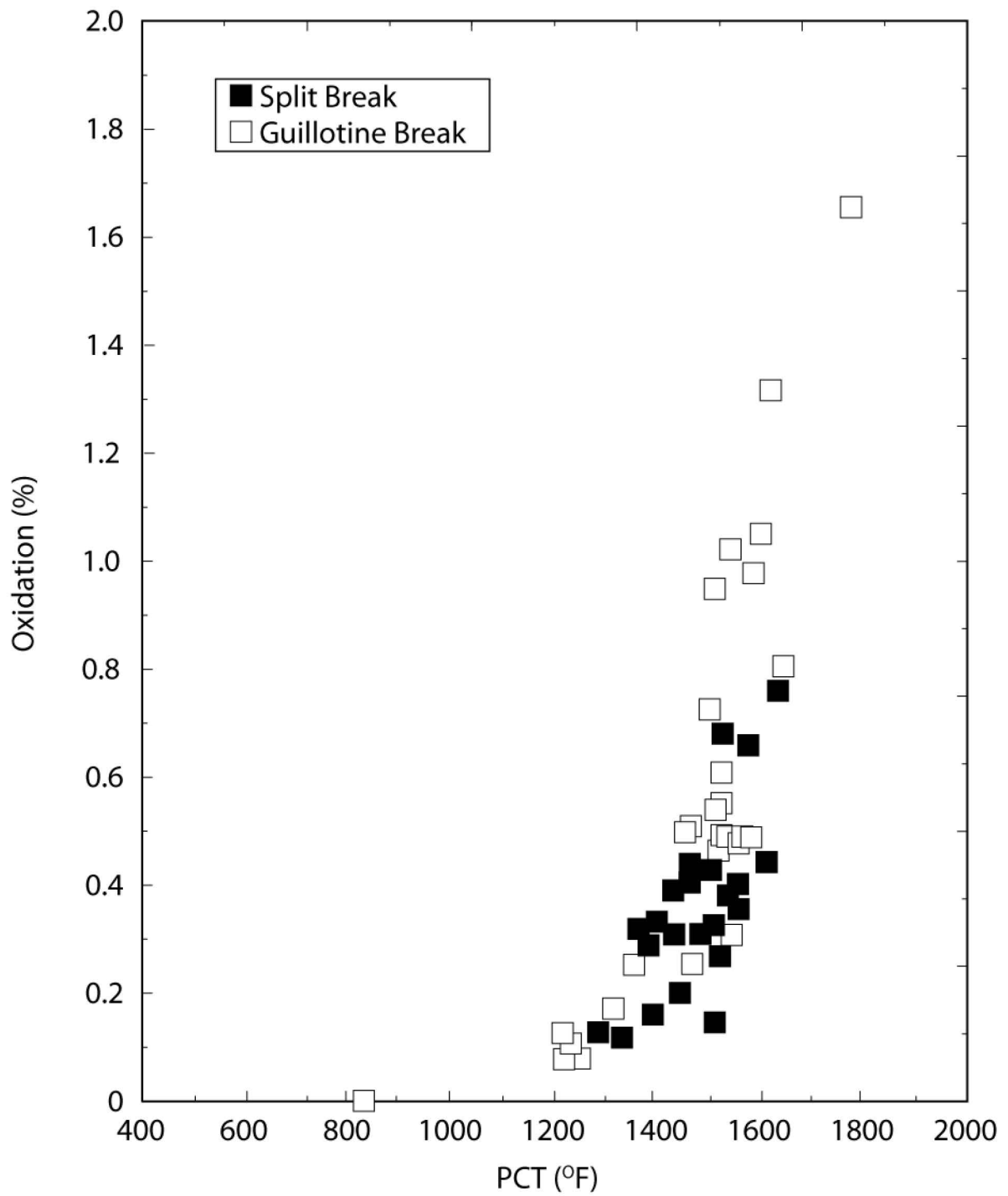


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

PCT VS ONE-SIDED BREAK AREA

FIGURE 15.4.1-3

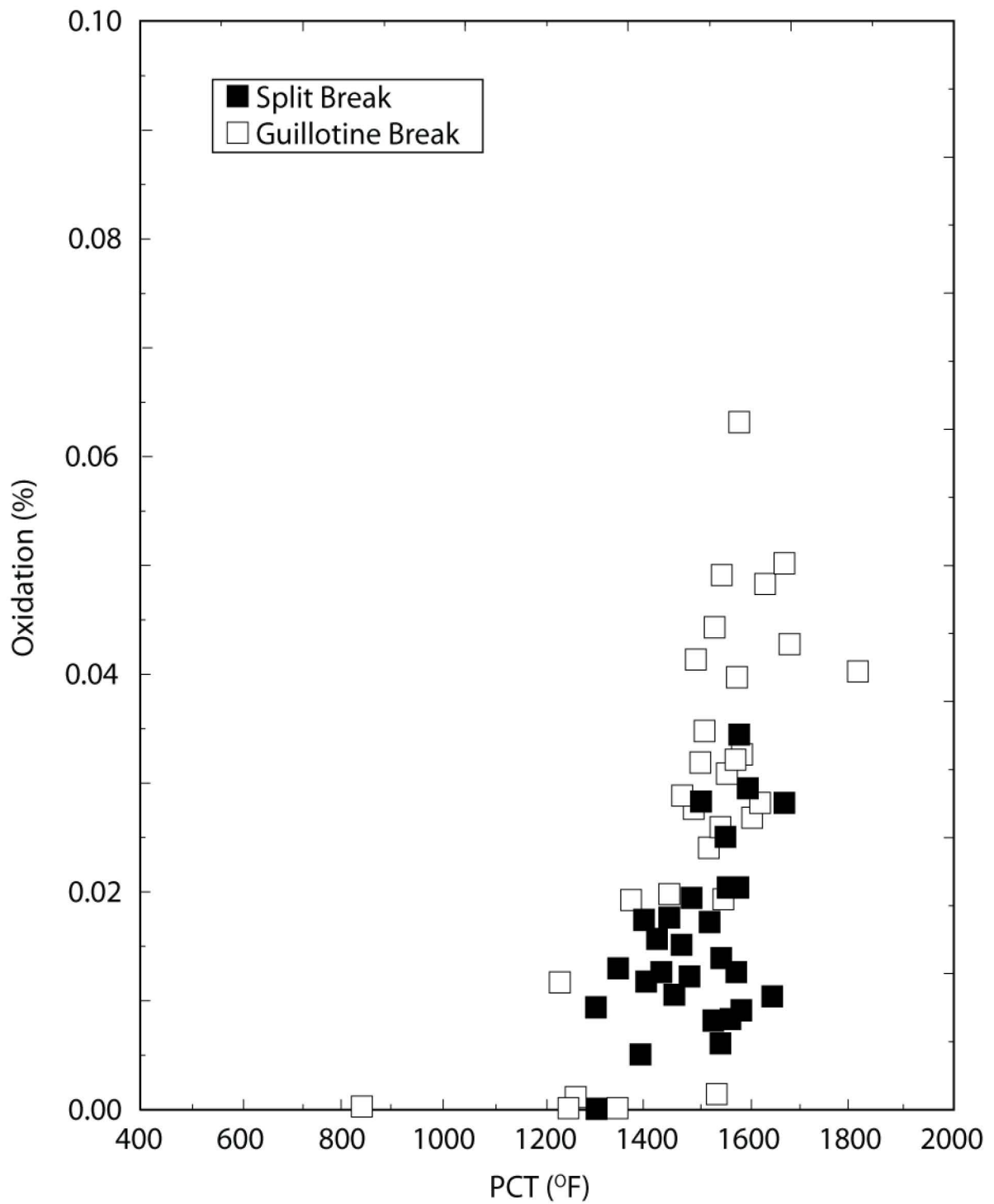


Amendment No. 27 (04/15)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAXIMUM OXIDATION VS PCT
SCATTER PLOT FROM 59 CALCULATIONS

FIGURE 15.4.1-4

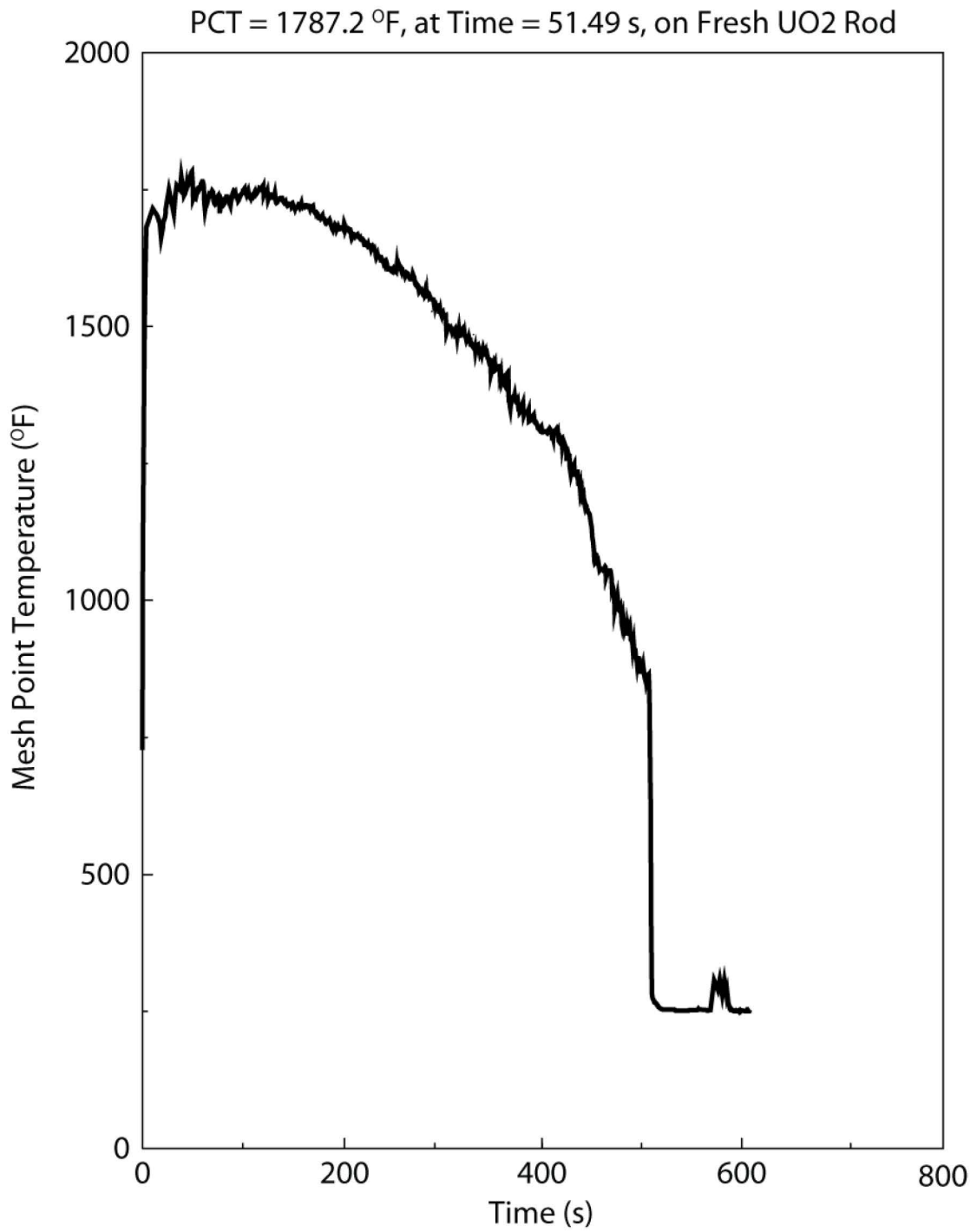


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**TOTAL OXIDATION VS PCT
SCATTER PLOT FROM 59 CALCULATIONS**

FIGURE 15.4.1-5

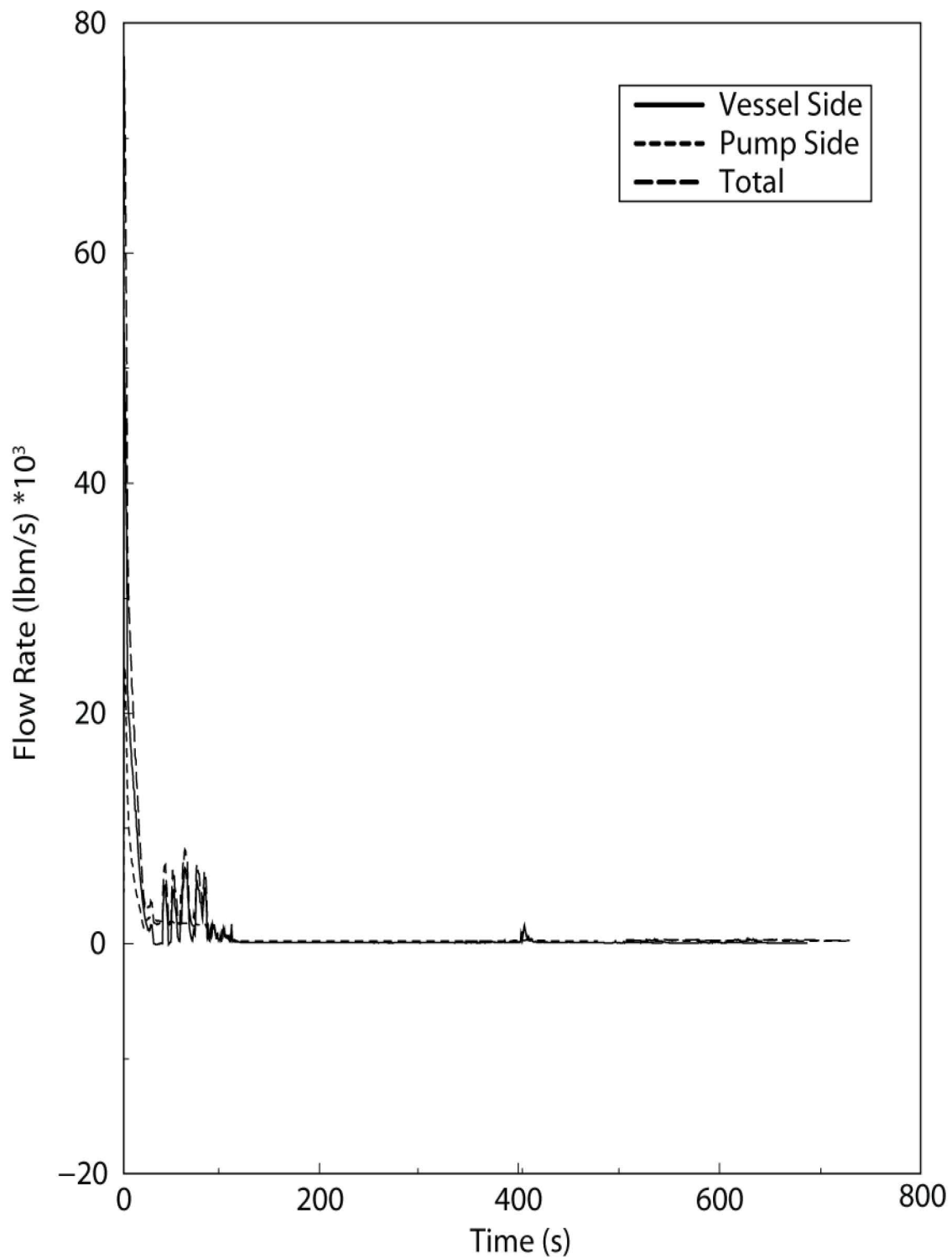


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**PEAK CLADDING TEMPERATURE
(INDEPENDENT OF ELEVATION) FOR THE
LIMITING CASE**

FIGURE 15.4.1-6

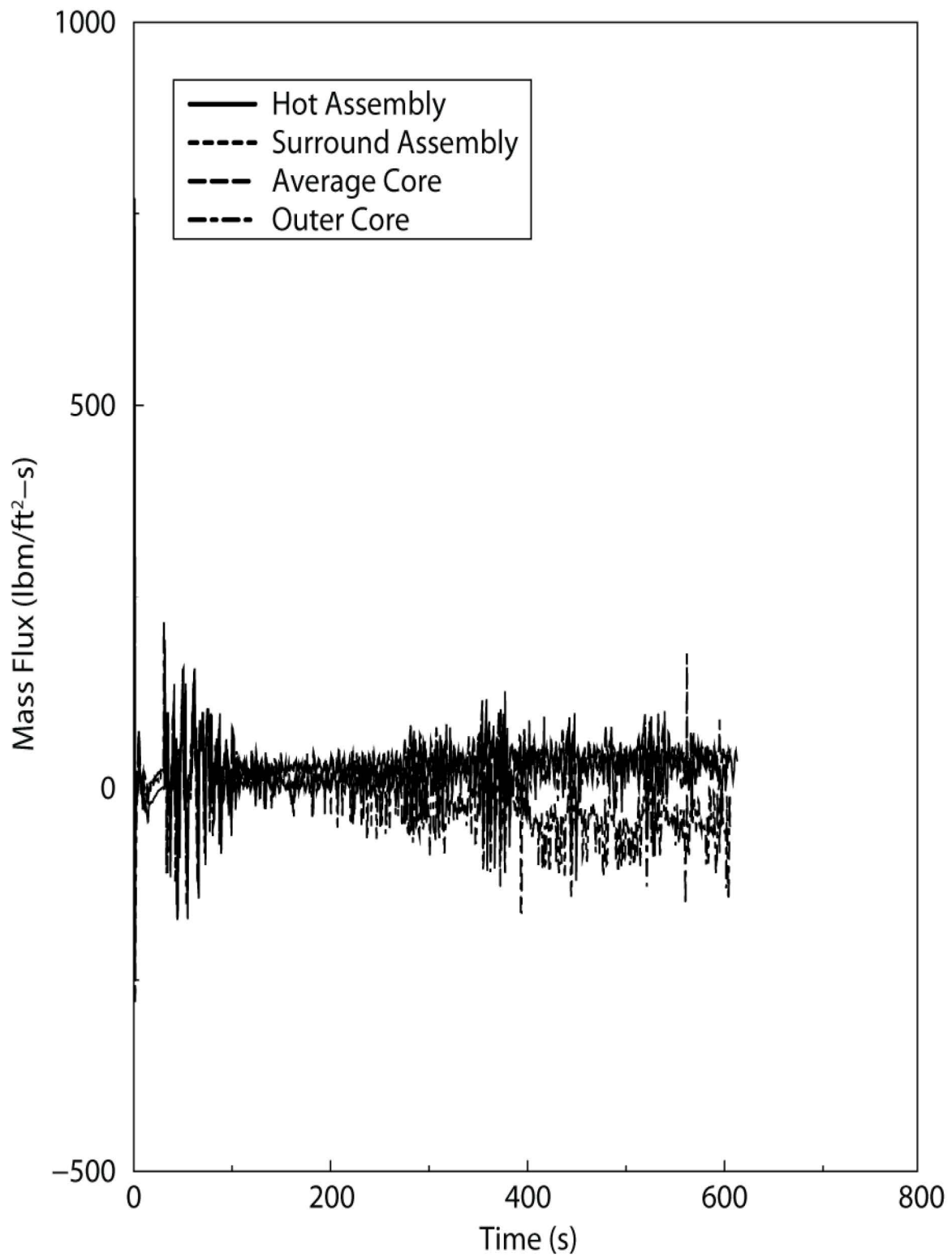


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

BREAK FLOW FOR THE LIMITING CASE

FIGURE 15.4.1-7

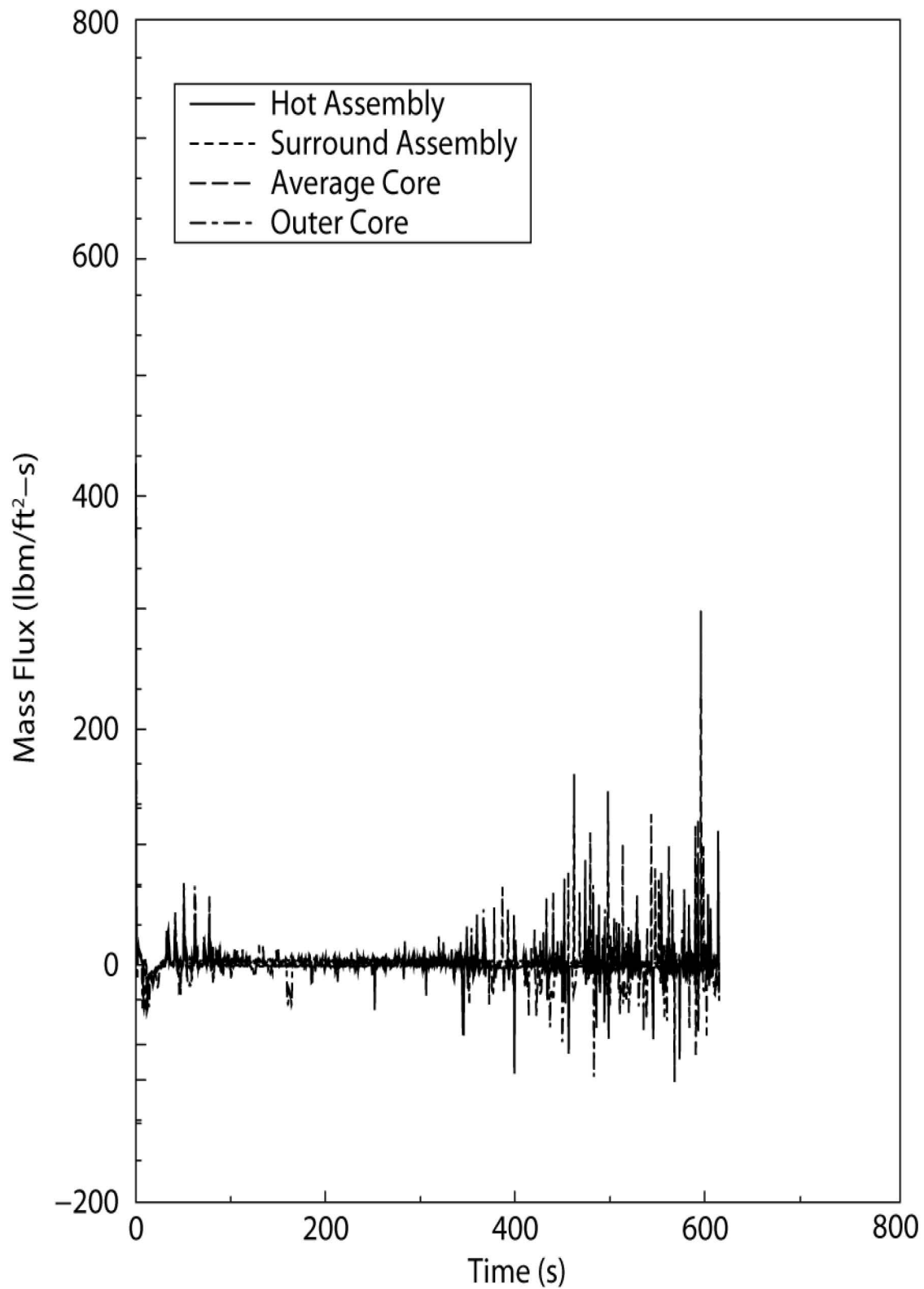


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**CORE INLET MASS FLUX FOR THE
LIMITING CASE**

FIGURE 15.4.1-8

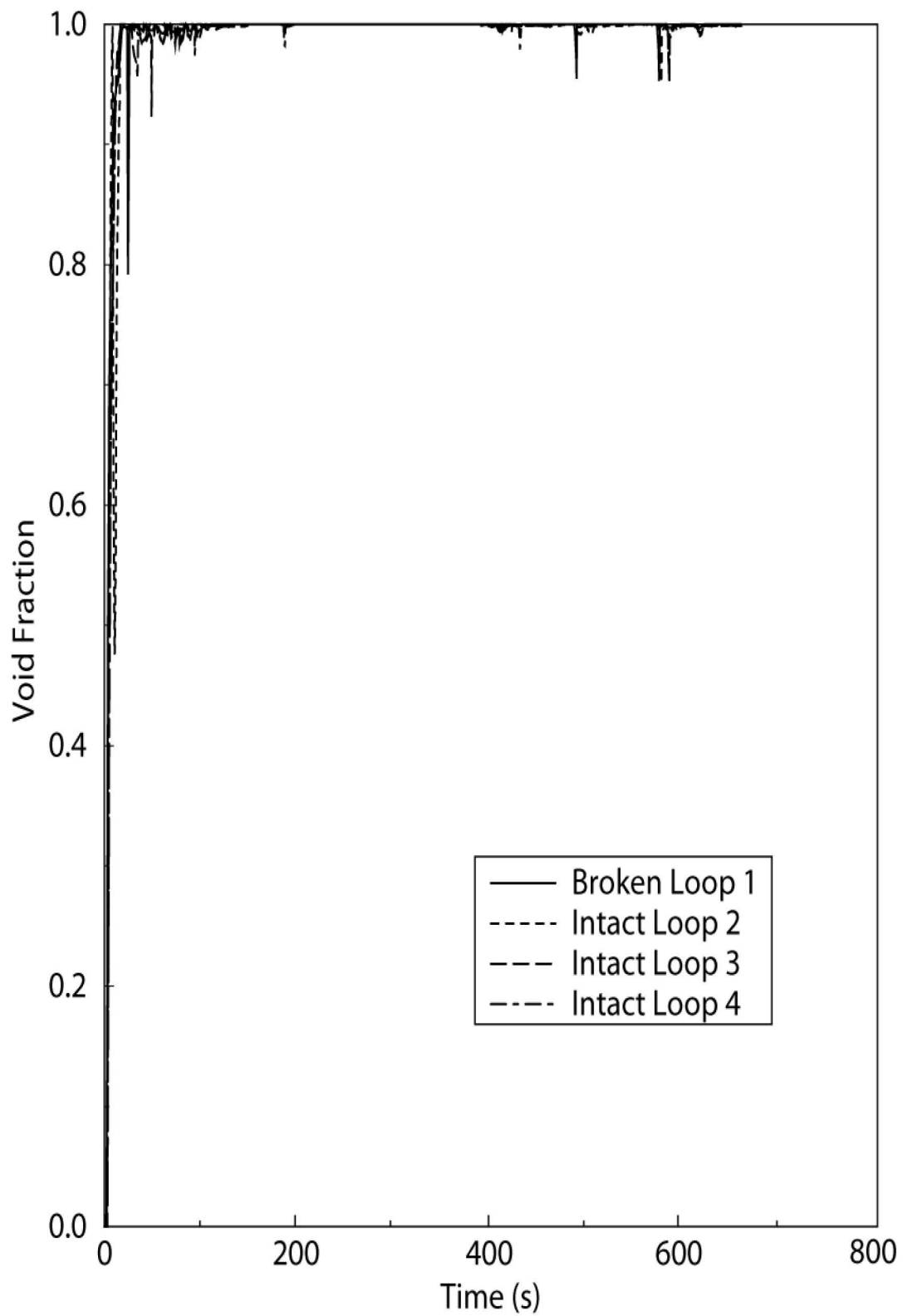


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**CORE OUTLET MASS FLUX FOR THE
LIMITING CASE**

FIGURE 15.4.1-9

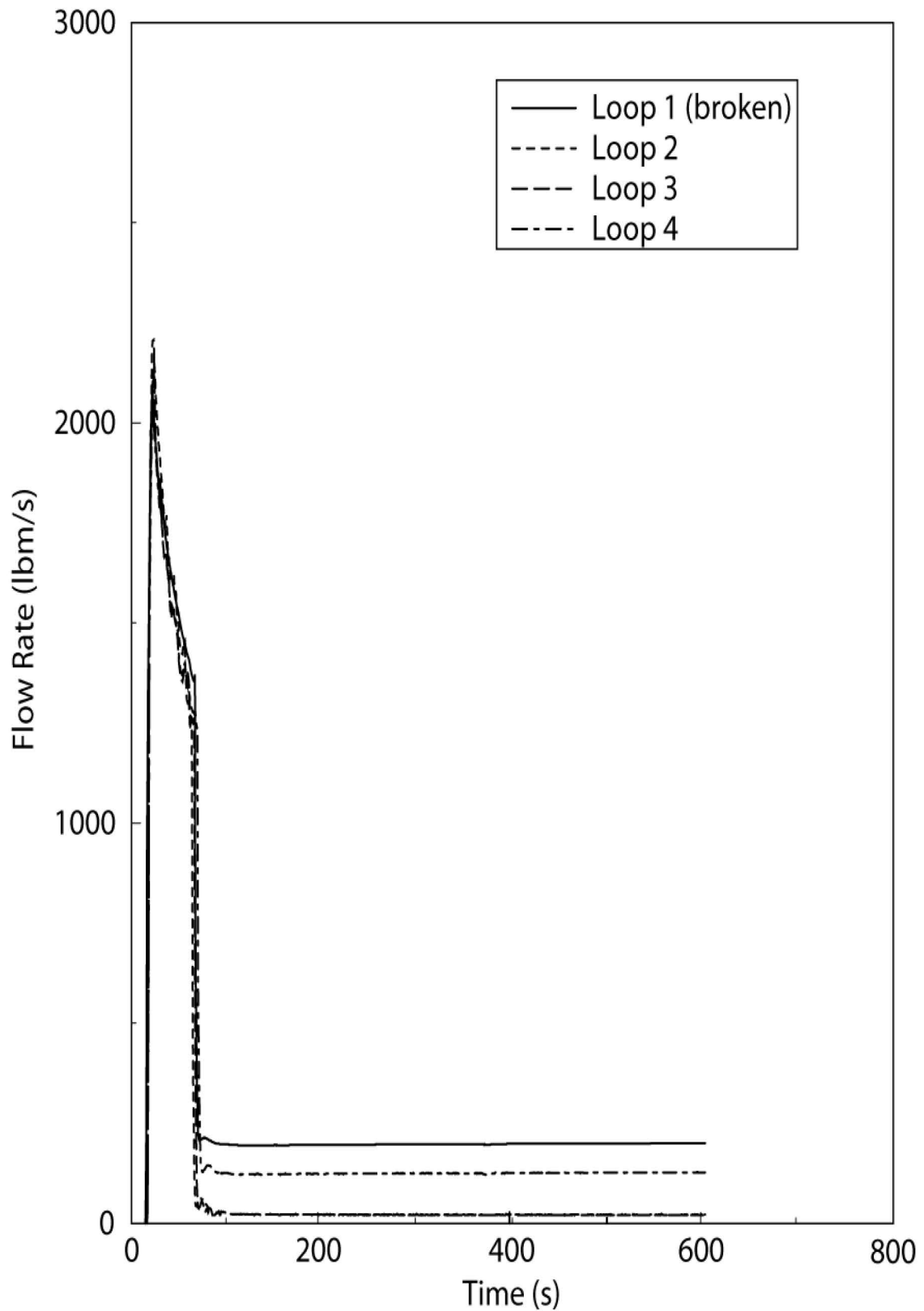


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**VOID FRACTION AT RCS PUMPS FOR THE
LIMITING CASE**

FIGURE 15.4.1-10

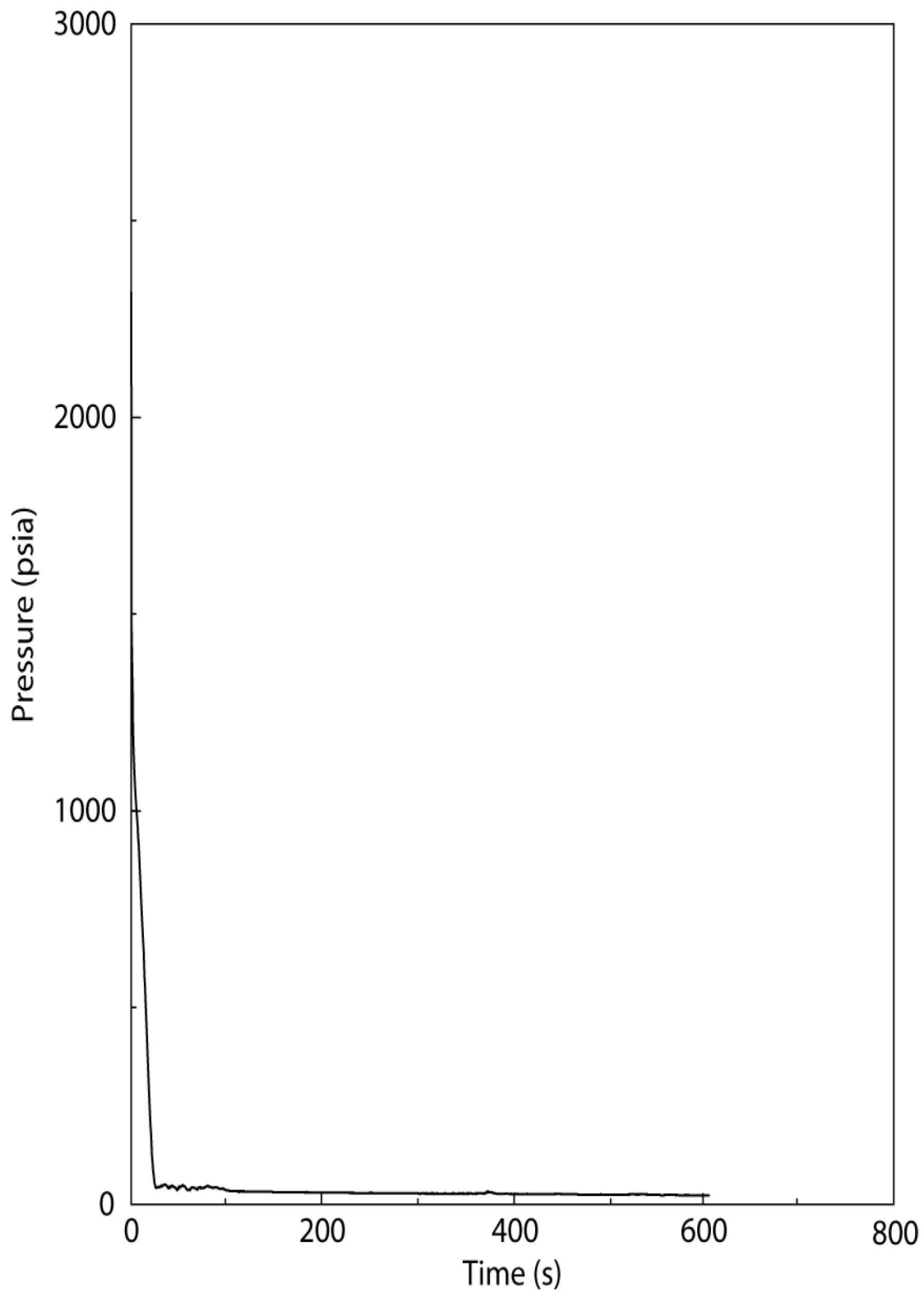


Amendment No. 27 (04/15)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

ECCS FLOW (INCLUDES SIT, LPSI AND
HPSI) FOR THE LIMITING CASE

FIGURE 15.4.1-11

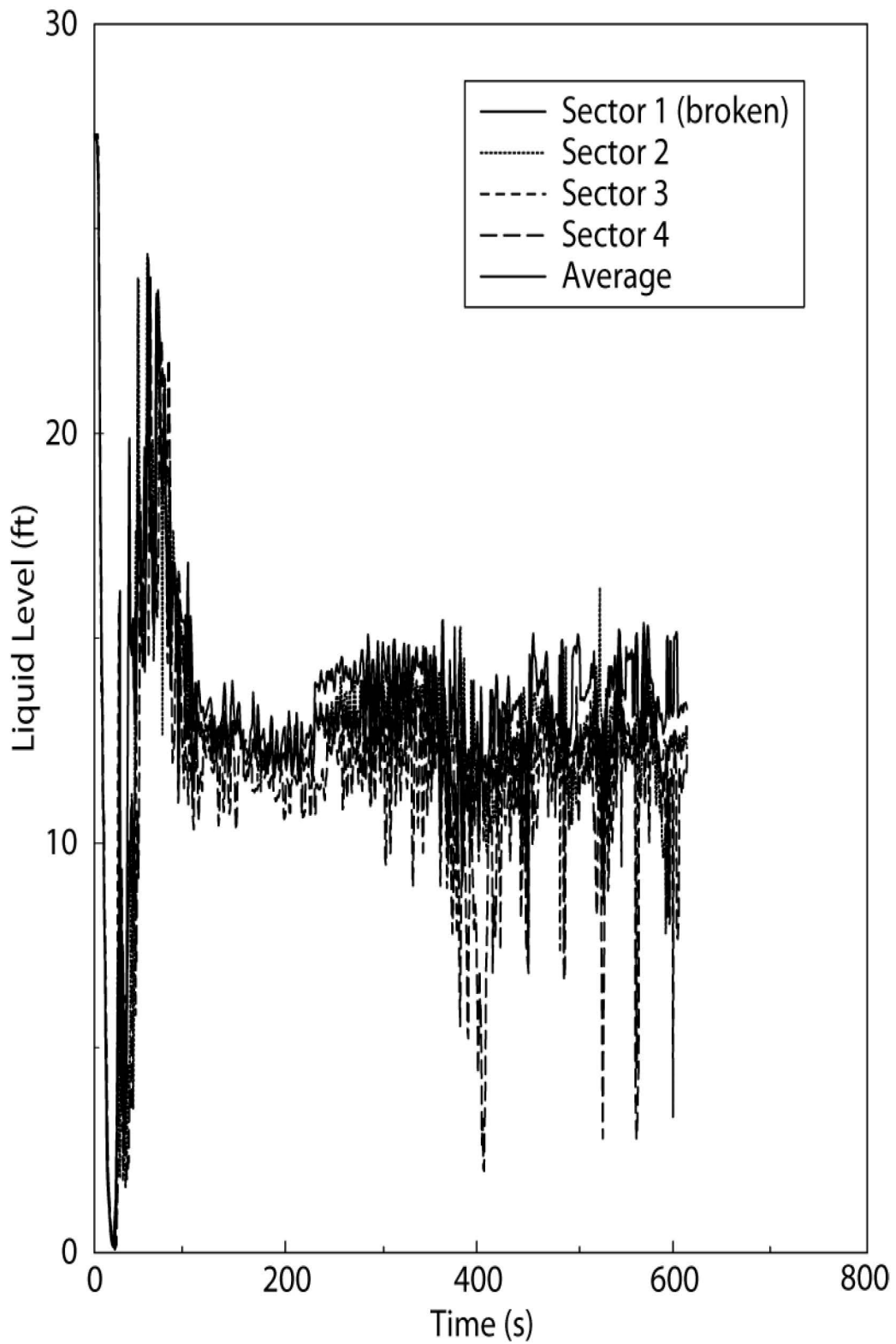


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**UPPER PLENUM PRESSURE FOR THE
LIMITING CASE**

FIGURE 15.4.1-12

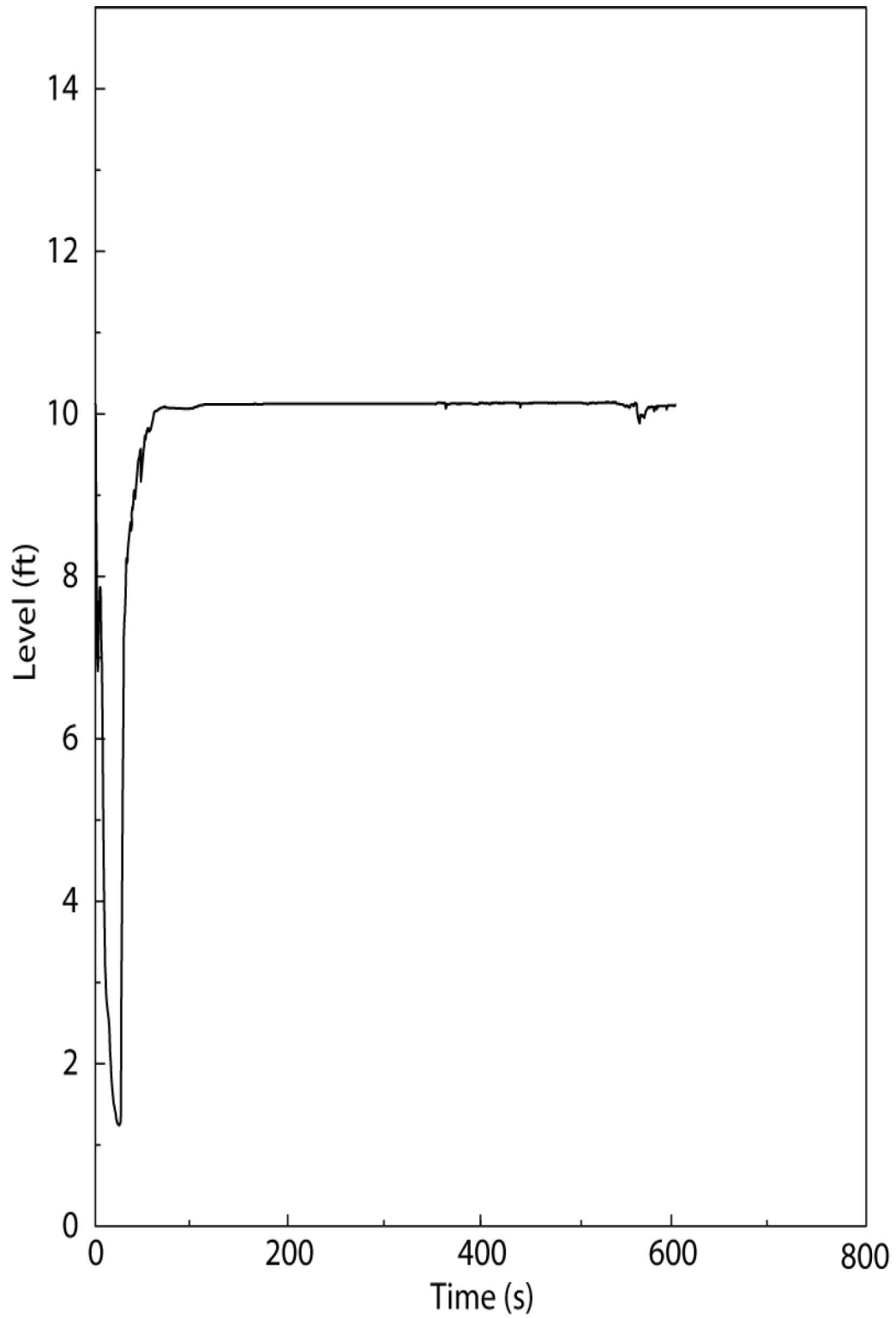


Amendment No. 27 (04/15)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

COLLAPSED LIQUID LEVEL IN THE
DOWNCOMER FOR THE LIMITING CASE

FIGURE 15.4.1-13

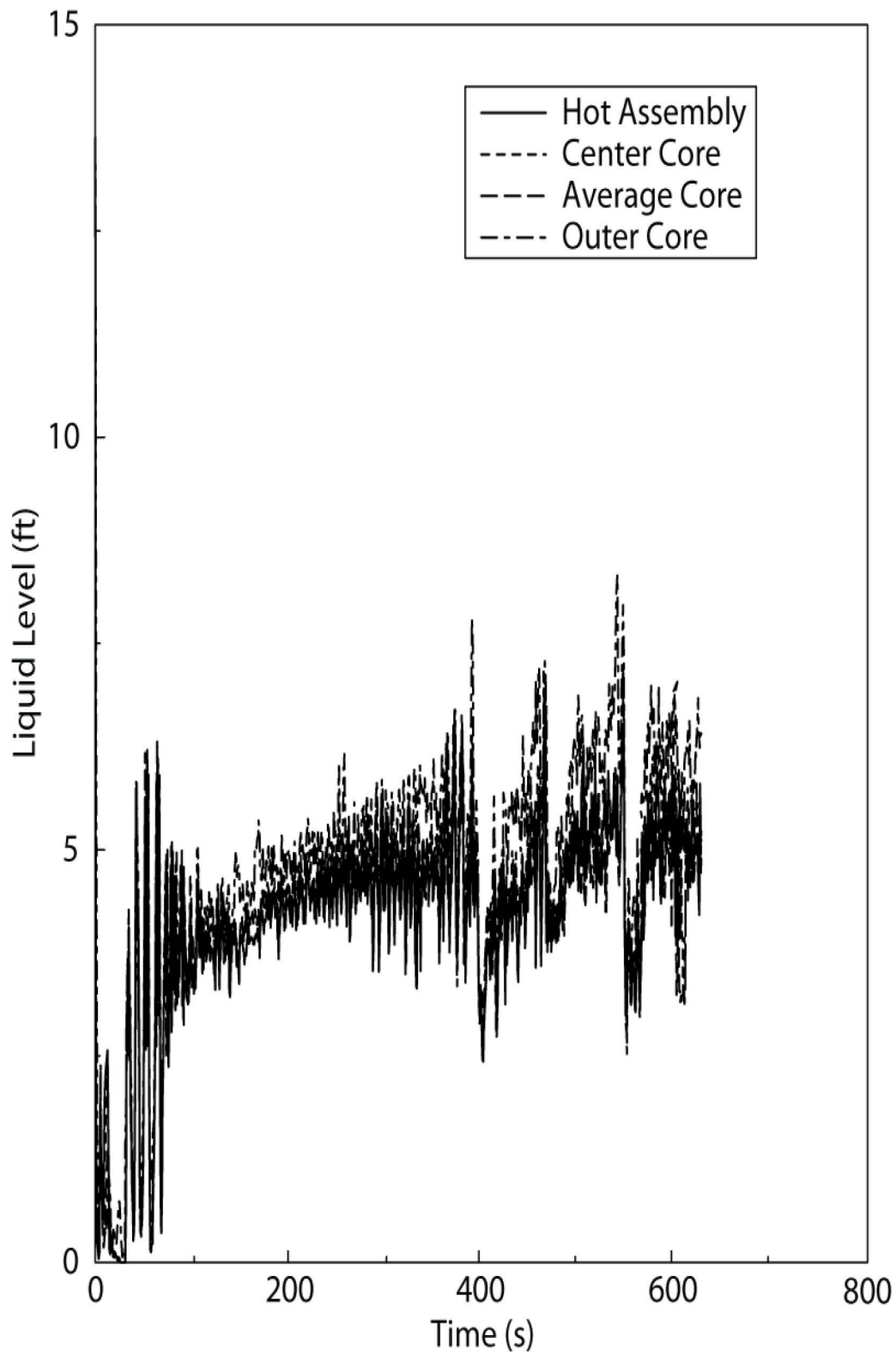


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**COLLAPSED LIQUID LEVEL IN THE
LOWER PLENUM FOR THE LIMITING
CASE**

FIGURE 15.4.1-14

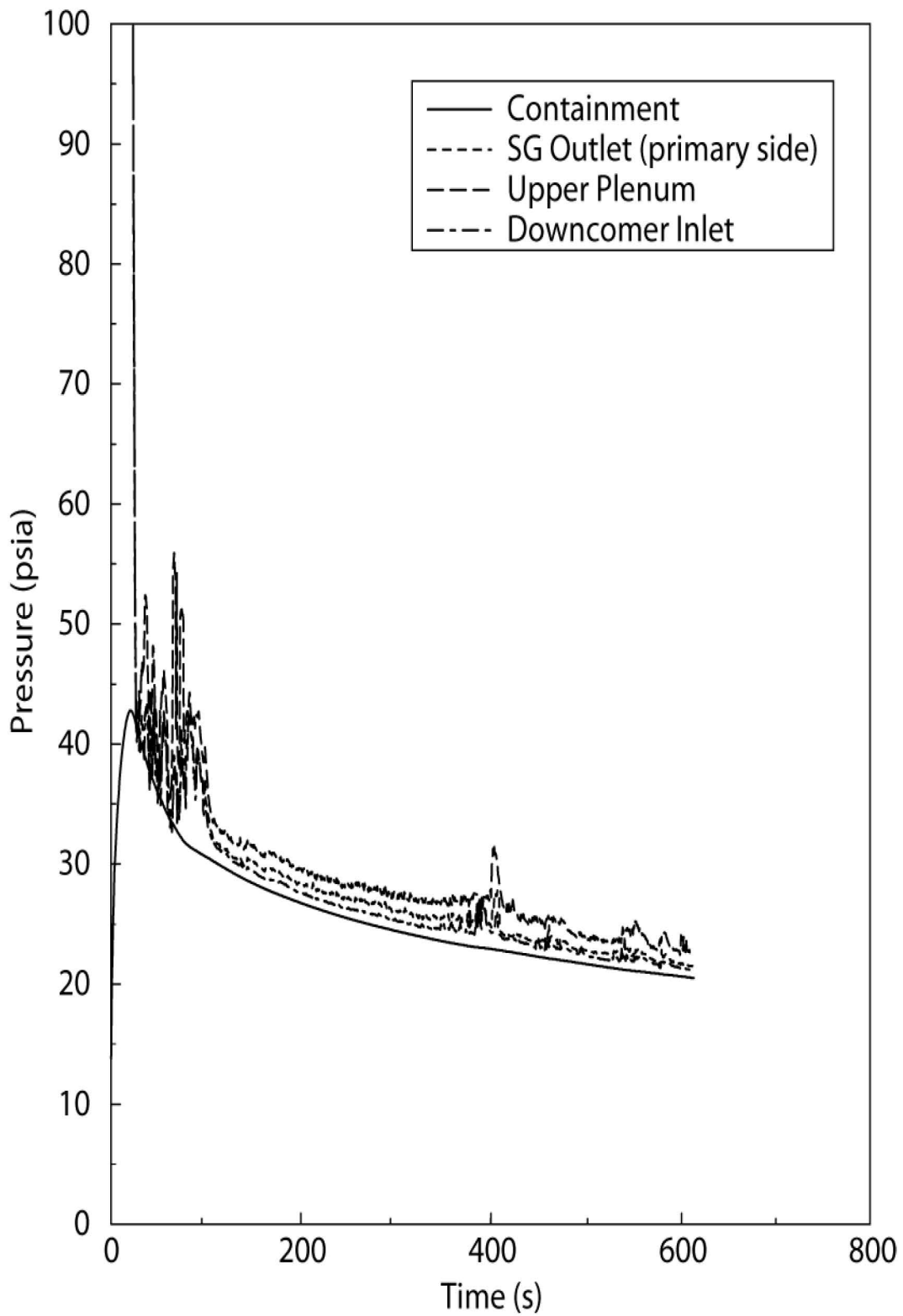


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**COLLAPSED LIQUID LEVEL IN THE CORE
FOR THE LIMITING CASE**

FIGURE 15.4.1-15

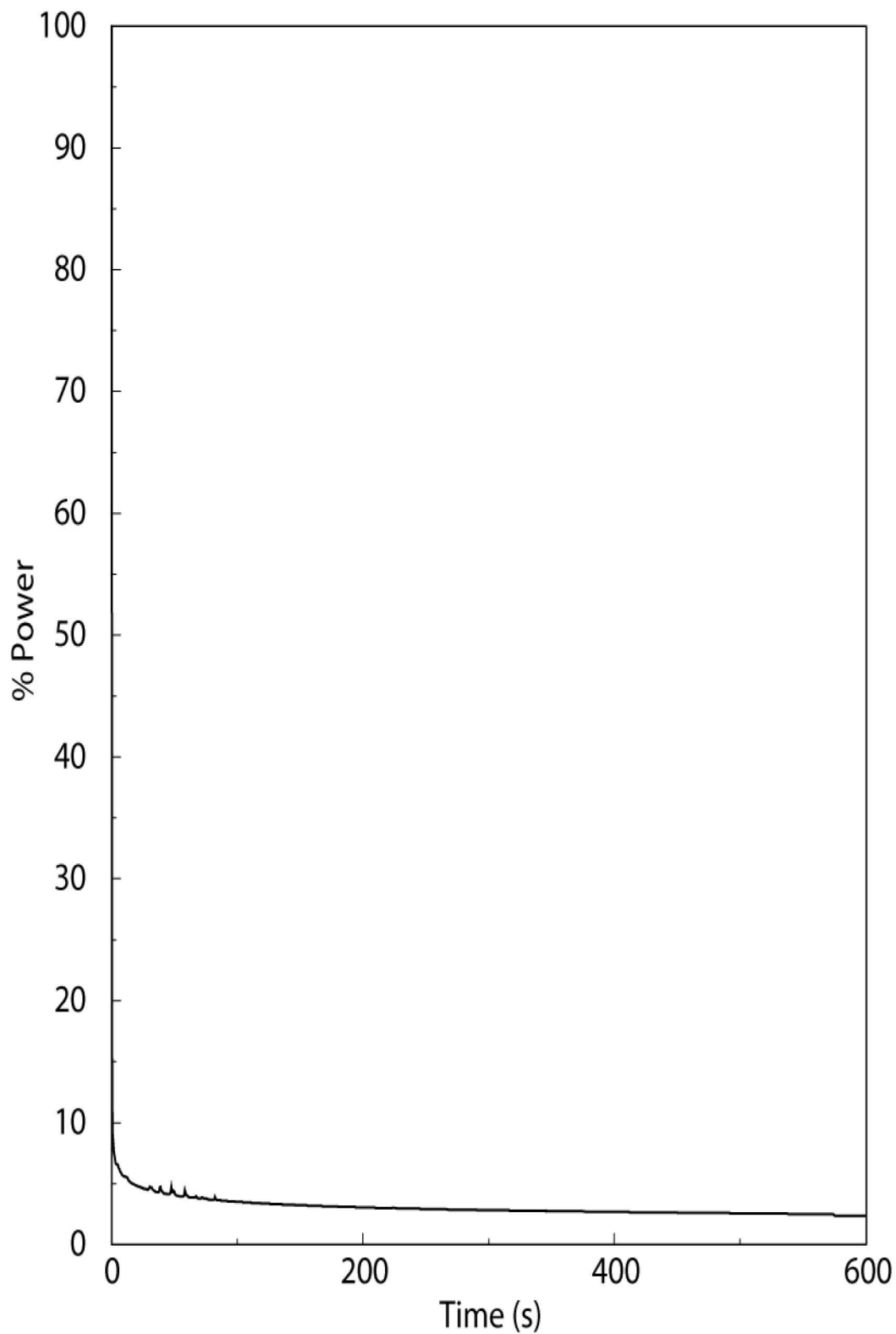


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**CONTAINMENT AND LOOP PRESSURES FOR
THE LIMITING CASE**

FIGURE 15.4.1-16

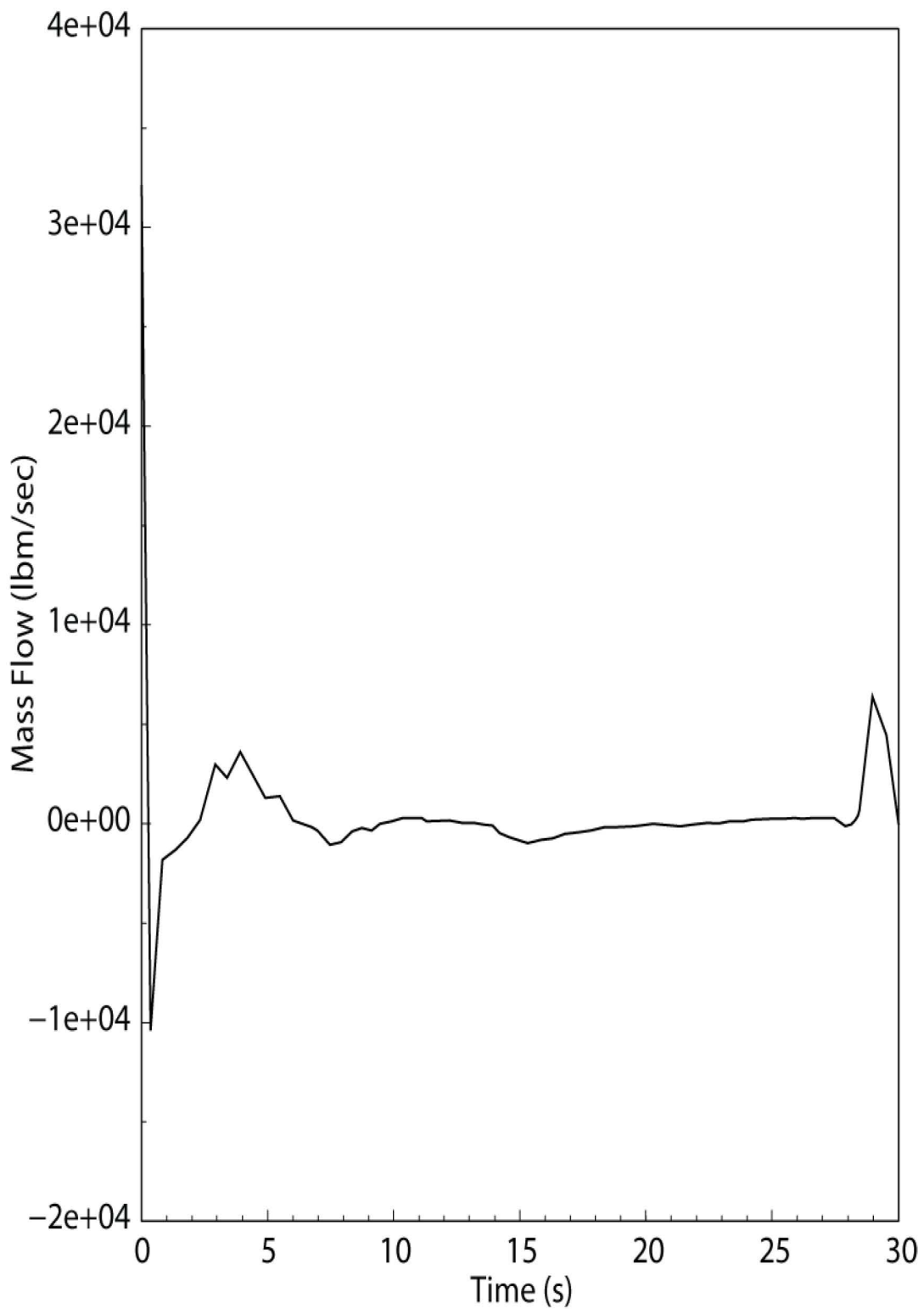


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**NORMALIZED POWER VS TIME FOR THE
LIMITING CASE**

FIGURE 15.4.1-17

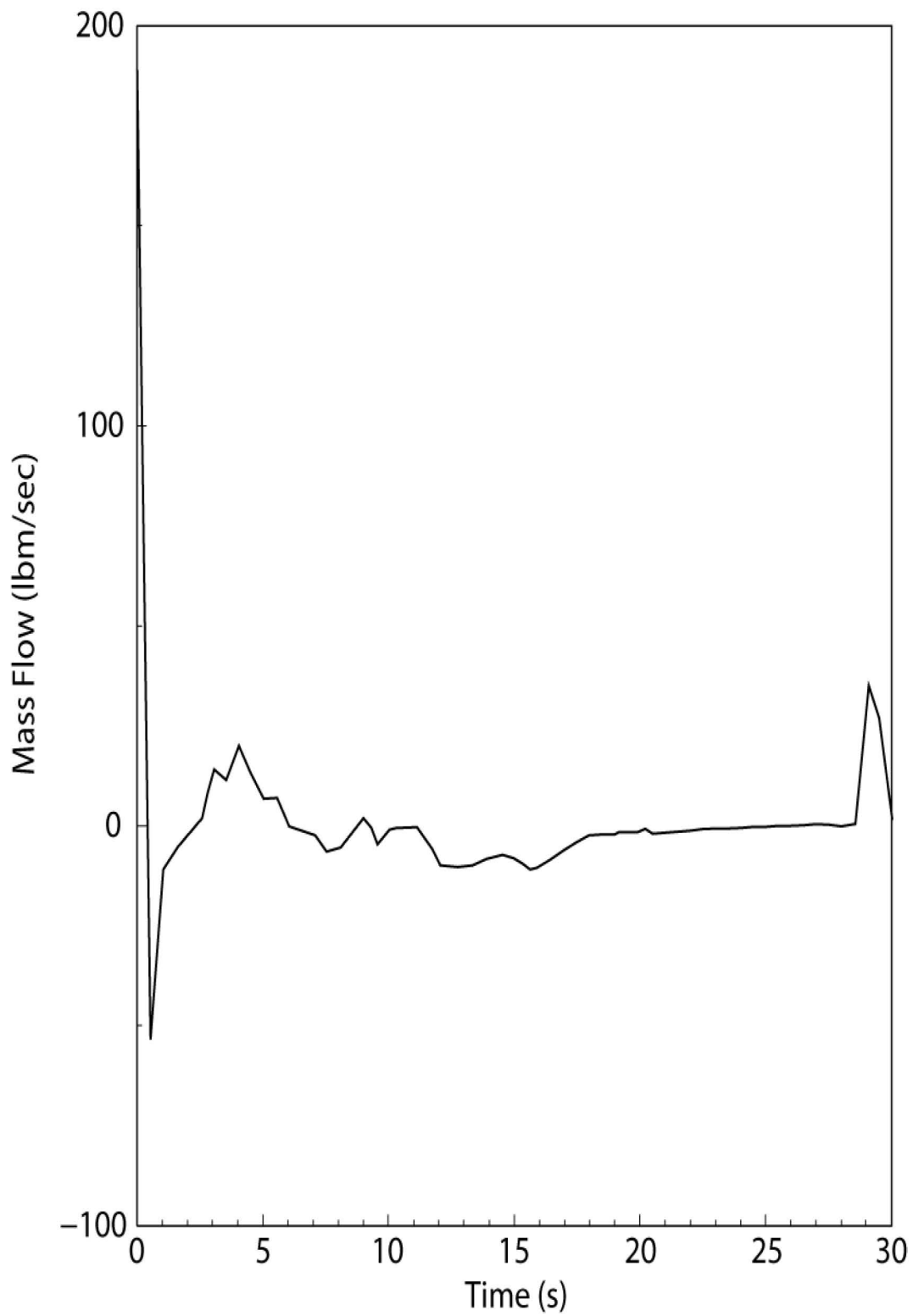


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**AVERAGE CORE INLET FLOW RATE DURING
BLOWDOWN (EOB = 25s) FOR THE LIMITING
CASE**

FIGURE 15.4.1-18

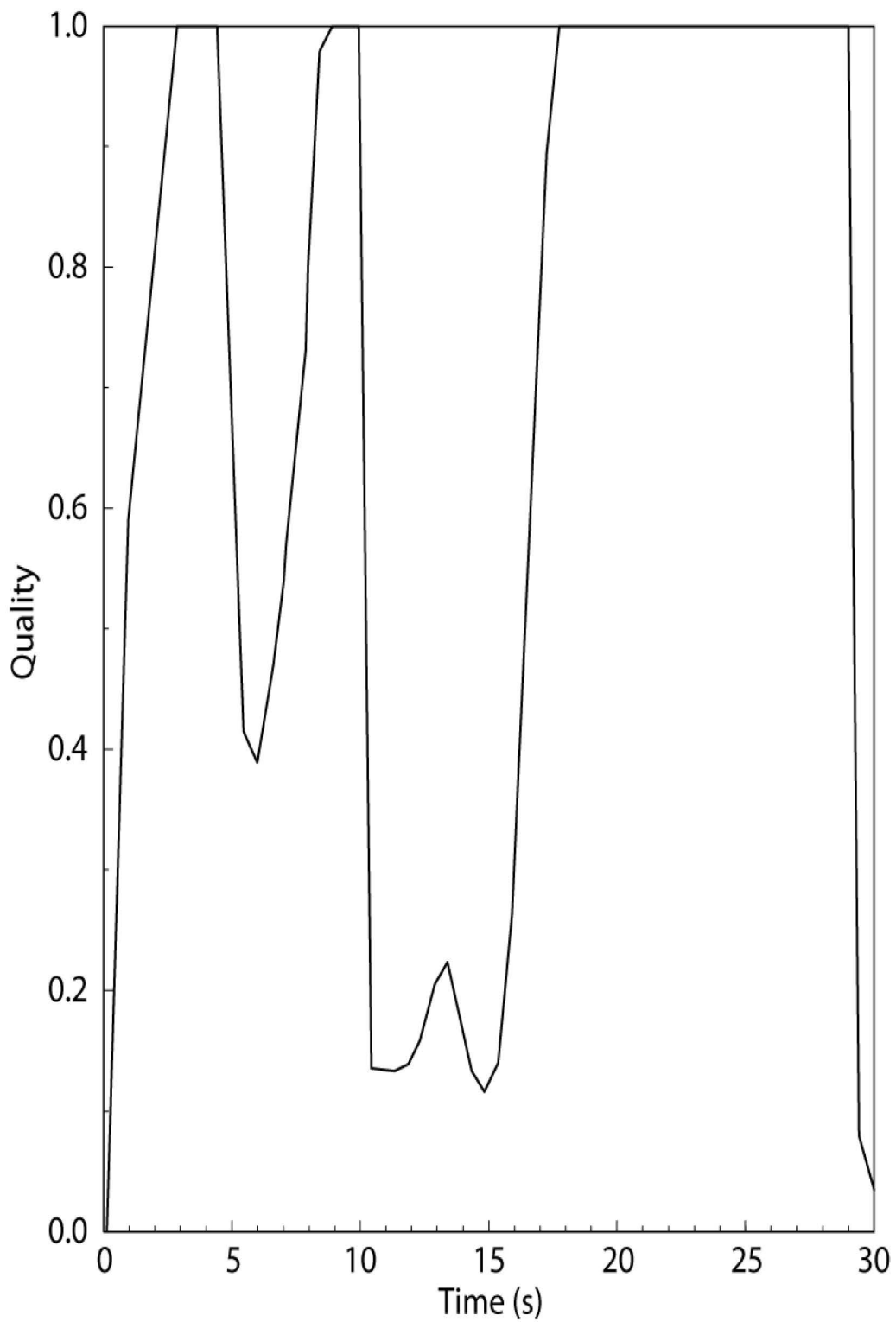


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**HOT CHANNEL INLET FLOW RATE DURING
BLOWDOWN (EOB = 25s) FOR THE LIMITING
CASE**

FIGURE 15.4.1-19

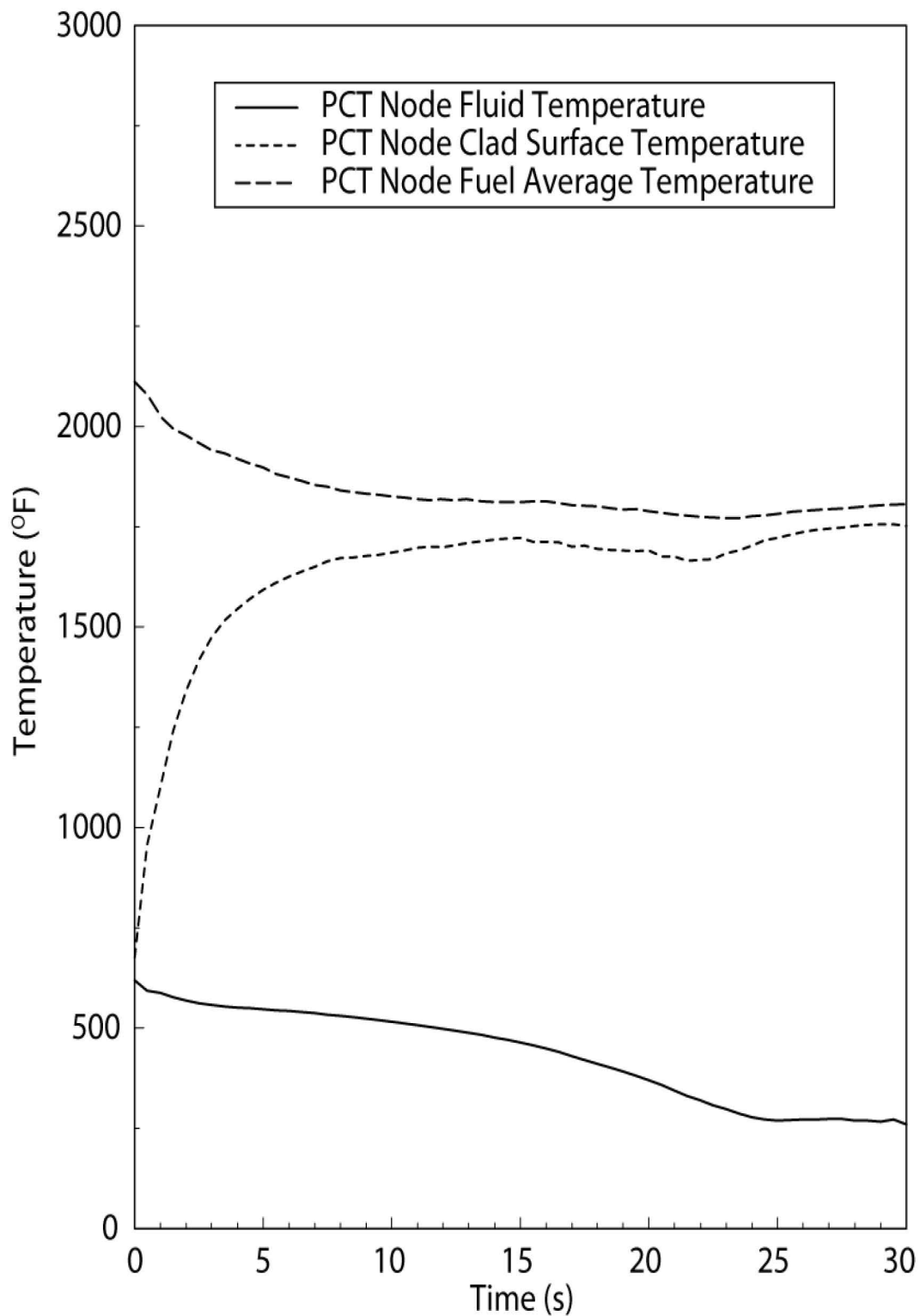


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**PCT NODE FLUID QUALITY DURING
BLOWDOWN (EOB = 25s) FOR THE LIMITING
CASE**

FIGURE 15.4.1-20

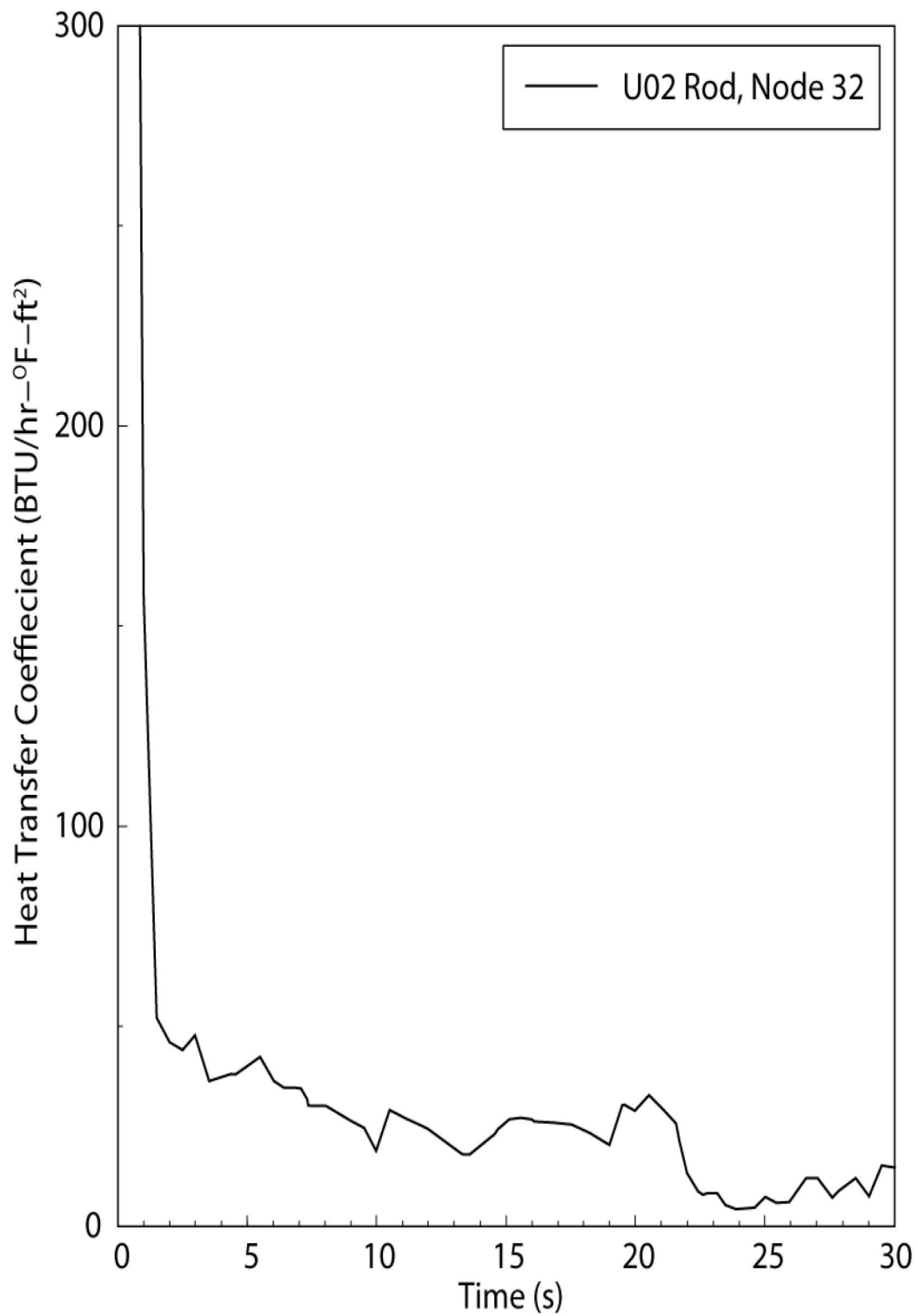


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

PCT NODE FUEL (AVERAGE), CLADDING, AND
FLUID TEMPERATURES DURING BLOWDOWN
(EOB = 25s) FOR THE LIMITING CASE

FIGURE 15.4.1-21

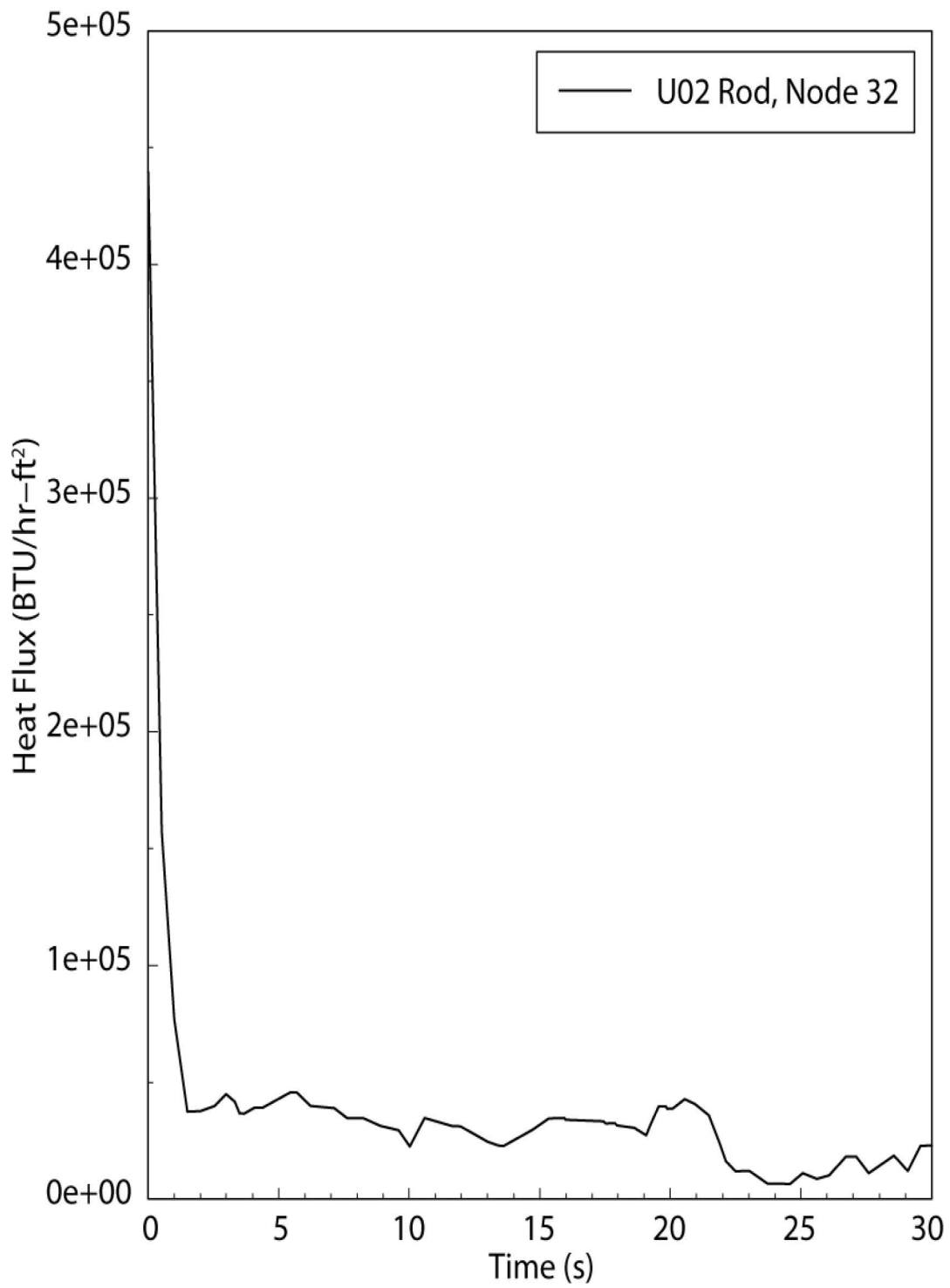


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**PCT NODE HEAT TRANSFER COEFFICIENT
DURING BLOWDOWN (EOB = 25s) FOR THE
LIMITING CASE**

FIGURE 15.4.1-22

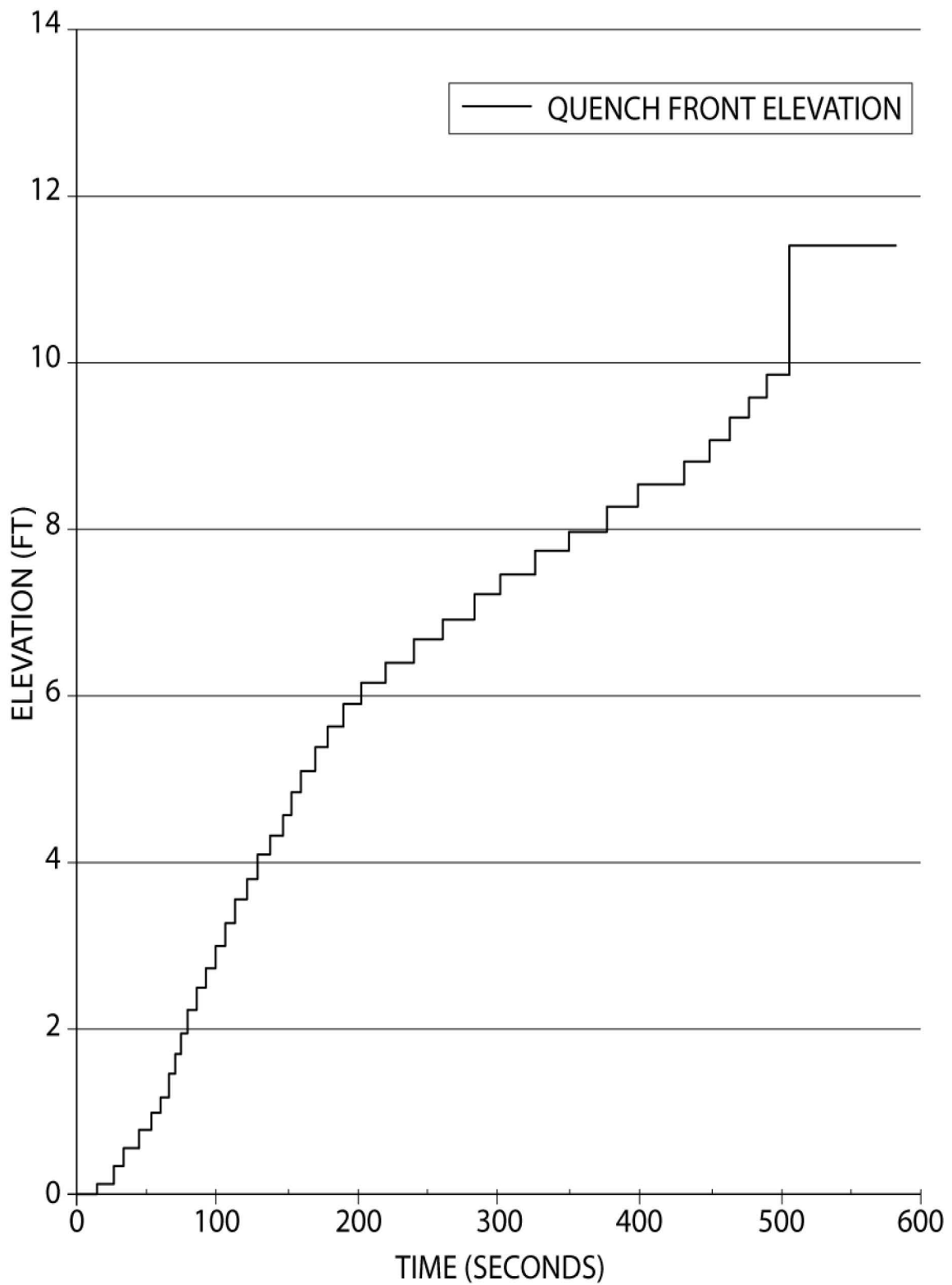


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**PCT NODE HEAT FLUX DURING BLOWDOWN
(EOB = 25s) FOR THE LIMITING CASE**

FIGURE 15.4.1-23

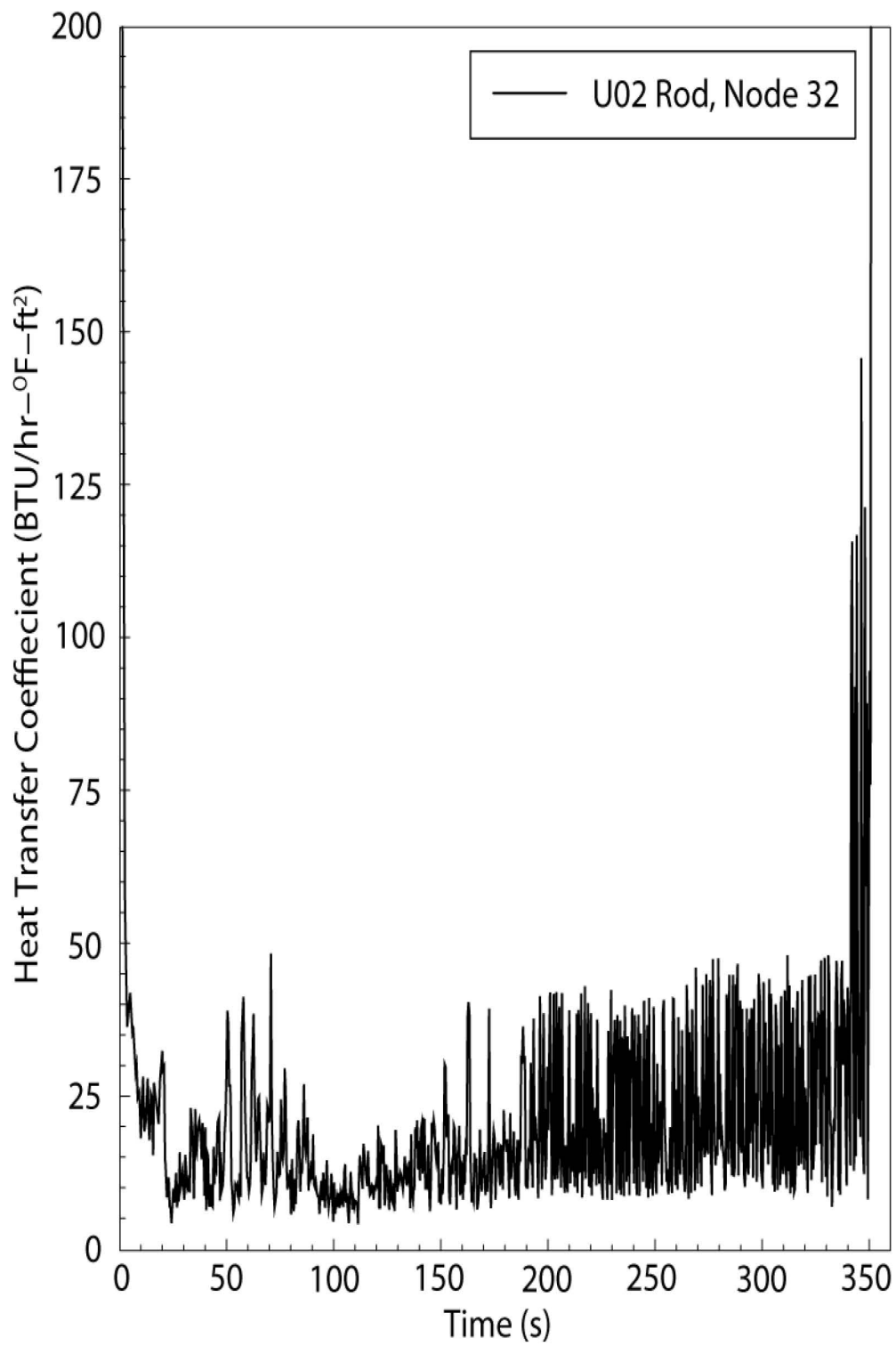


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**CORE QUENCH LEVEL FOR THE LIMITING
CASE**

FIGURE 15.4.1-24

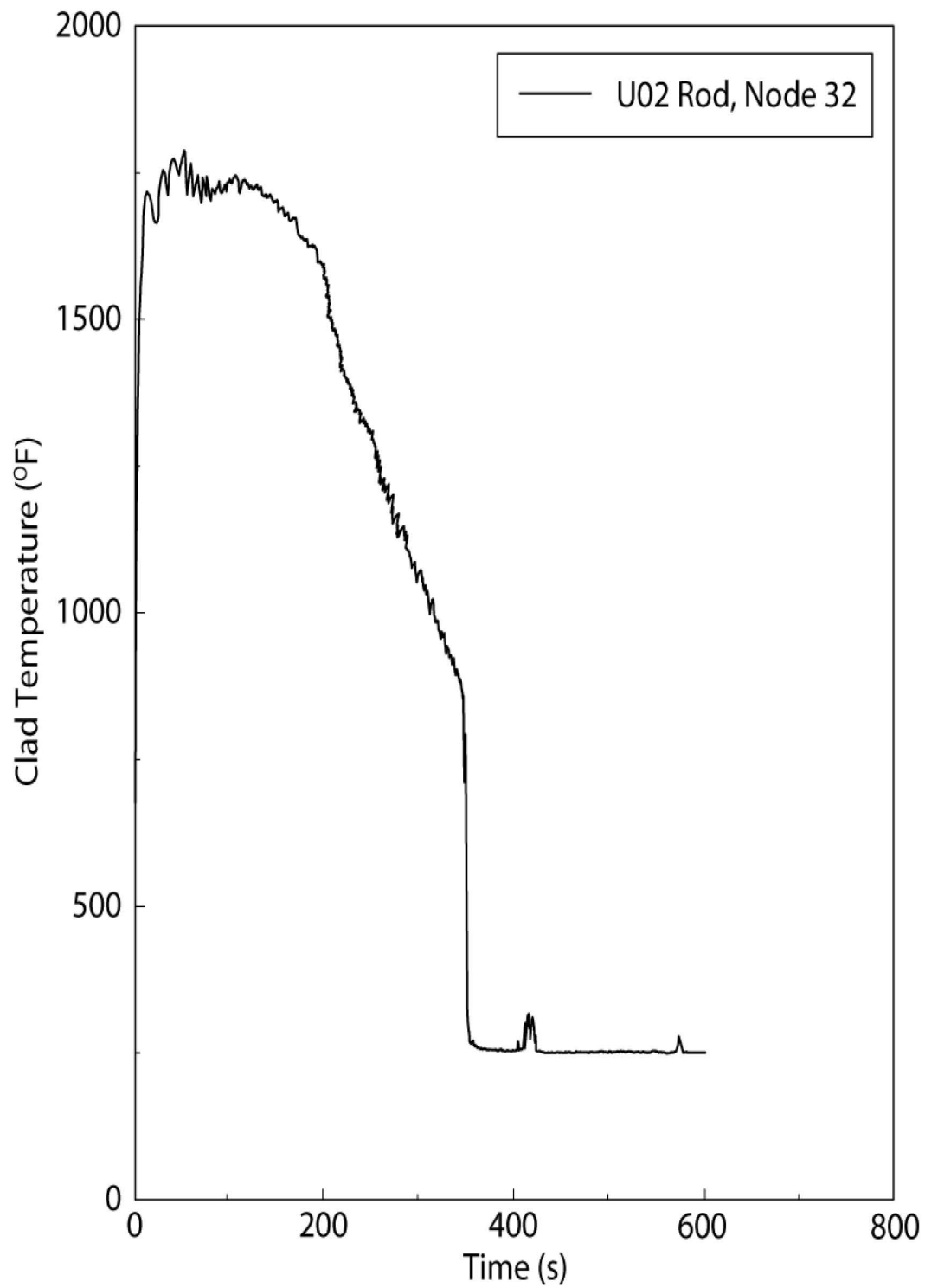


Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**PCT NODE HEAT TRANSFER COEFFICIENT
(TO PCT NODE QUENCH) FOR THE LIMITING
CASE**

FIGURE 15.4.1-25

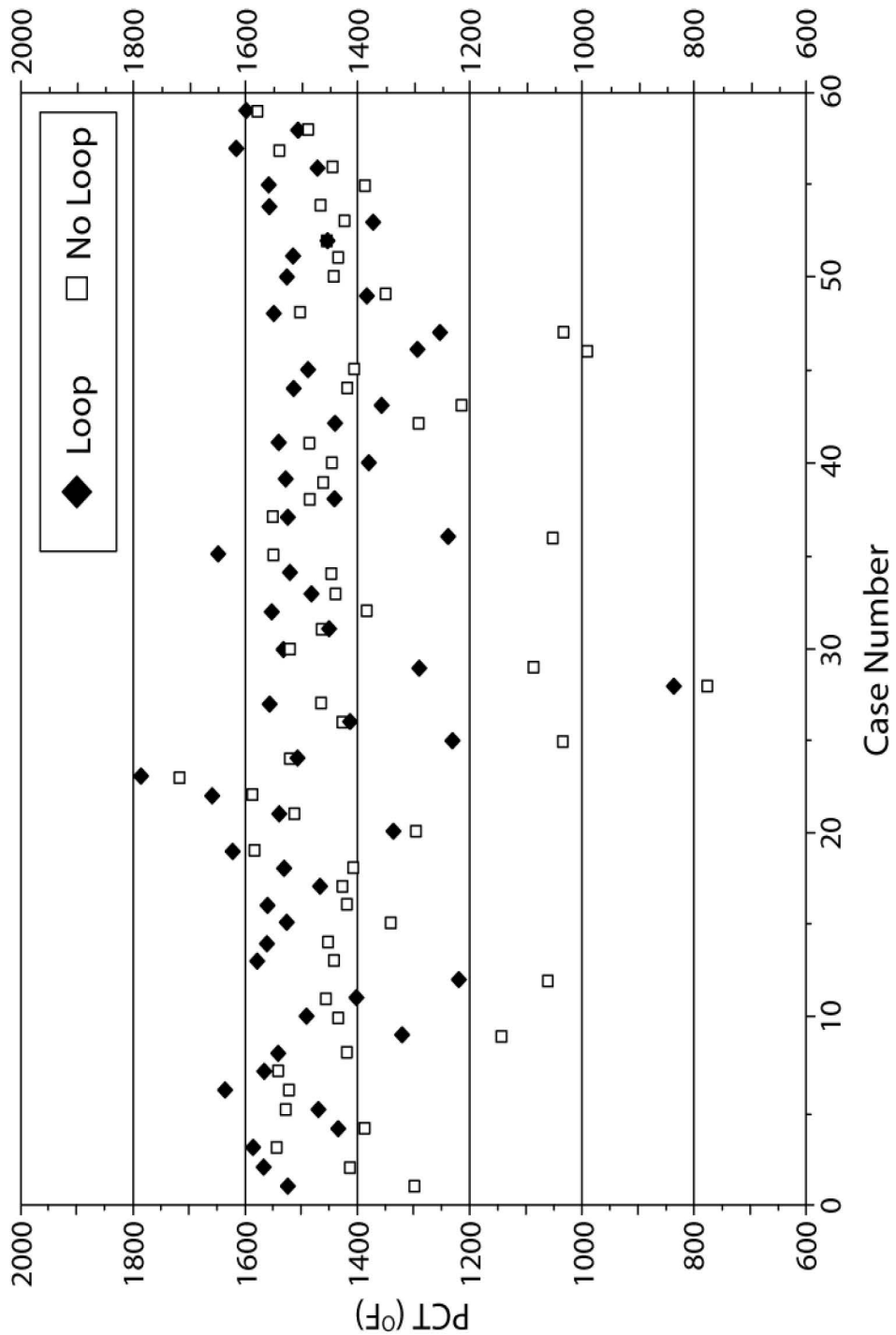


Amendment No. 27(04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**PCT NODE CLADDING TEMPERATURE FOR
THE LIMITING CASE**

FIGURE 15.4.1-26



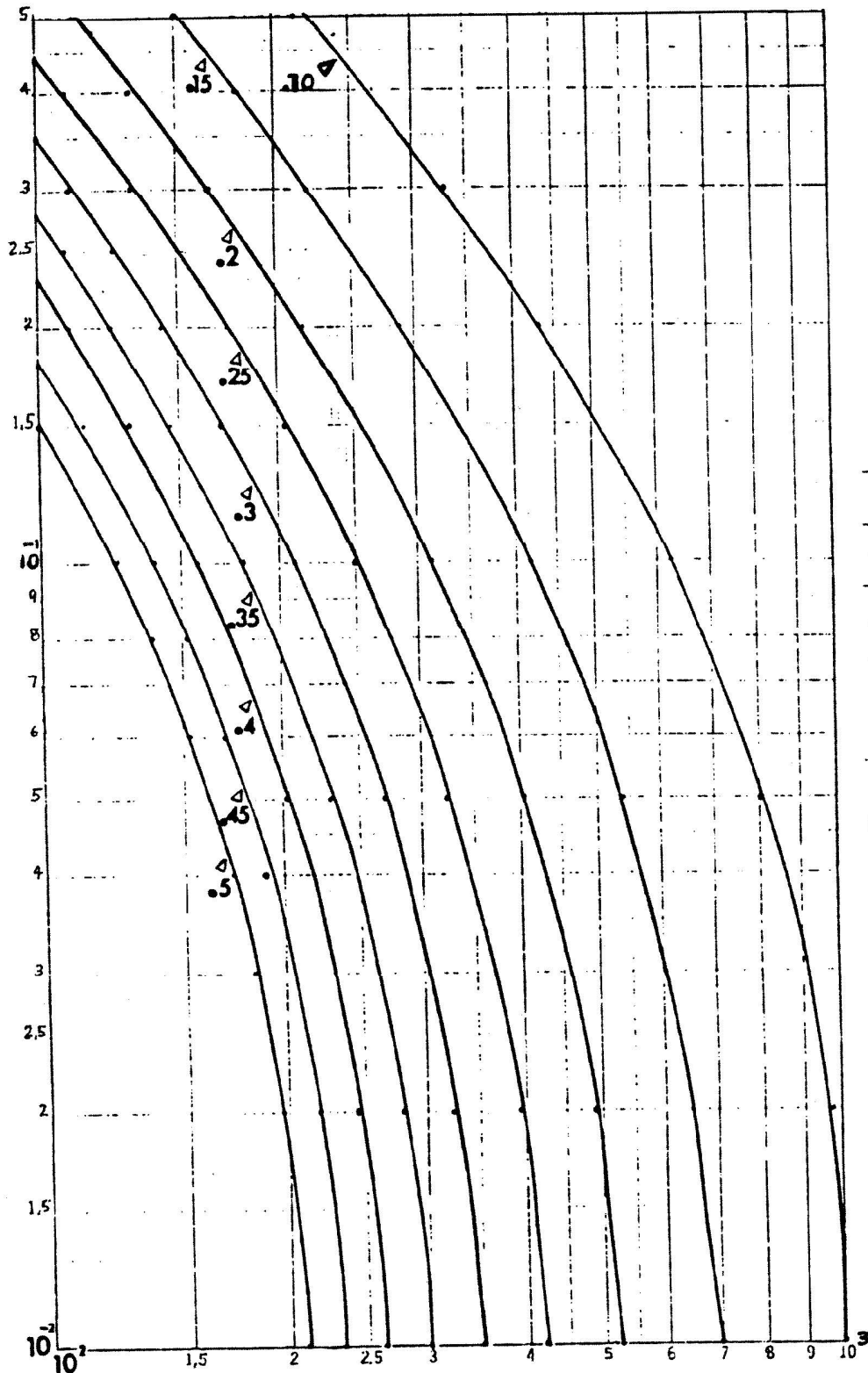
Amendment No. 27 (04/15)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

GDC 35 LOOP VS NO-LOOP CASES

FIGURE 15.4.1-27

BYPASS LEAKAGE (FRACTION OF CONTAINMENT LEAK RATE)



Note: Curves define combinations of conditions for which post LOCA control room doses are within GDC 19.

AMENDMENT NO. 11 (7/92)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1
ALLOWABLE CONTROL ROOM INTAKE AS A
FUNCTION OF CONT. AND BYPASS LEAKAGE
FIGURE 15.4.1-28

15.4.2 WASTE GAS DECAY TANK LEAKAGE OR RUPTURE

15.4.2.1 Identification of Causes

A release of the stored radioactive gases in the waste gas decay tanks in the gaseous waste system described in Section 11.3 as a result of component failure or inadvertent venting is considered improbable. The waste gas decay tanks are not subjected to high pressures or unusual stresses. Inadvertent release of the contents of the waste gas decay tanks would be detected by the radiation detectors in the line leading to the plant vent. In the event of a high radiation level of the effluent gases to the vent that would result in exceeding the limits of 10 CFR 20, an alarm will sound in the control room and the flow of gas to the vent will be automatically interrupted.

The radioactive gases stored in the decay tanks consist of fission gases and the hydrogen and nitrogen cover gases. The nitrogen is added in the various collection and holdup tanks to preclude the possibility of obtaining a flammable mixture of hydrogen gas. Hence, tank rupture as a result of ignition of hydrogen in the decay tank is remote.

To determine the upper limit to the radiological consequences resulting from a failure of a waste gas decay tank it is assumed that the contents of the tank are released instantaneously at ground level. The tank is assumed to contain fission gases stripped from one complete system volume of reactor coolant.

15.4.2.2 Radiological Analysis

15.4.2.2.1 Applicable Regulatory Guidance

This analysis is performed using the AST consistent with Regulatory Guide 1.183. In addition, recent NRC guidance given in Regulatory Issue Summary, RIS-2006-04, regarding application of the AST to WGDT events is also followed. The RIS-2006-04 guidance specifically endorses BTP 11-5, Rev. 3 from the Standard Review Plan, NUREG-0800.

15.4.2.2.2 Source Term

The source term released during the failure of a WGDT is comprised of 100% of the equilibrium noble gas activity of the RCS. The source term was derived from the St. Lucie specific core source term and 1% fission product release to coolant inventories using the ORIGEN 2.1 computer code. ORIGEN is an NRC approved methodology for source term development. The resulting WGDT source term is provided in Table 15.4.2-2.

15.4.2.2.3 Acceptance Criteria

The offsite dose acceptance criterion for a WGDT rupture accident of 0.1 rem total effective dose equivalent (TEDE), as specified in BTP 11-5, is chosen to apply to receptors located at the exclusion area boundary (EAB) and the low population zone (LPZ). As noted in UFSAR Section 11.3, the gaseous waste management system is designed to prevent an explosive gas mixture and the WGDTs are seismically designed. Per BTP 11-5, a dose limit of 2.5 rem could be applied. However, the more restrictive dose limit of 0.1 rem TEDE for systems not designed to withstand explosions and earthquakes was applied as a conservative way to establish a restrictive technical specification limit for the contents of the WGDTs. Note that this dose limit applies to the duration of the event (30 days), unlike most AST analyses of doses at the Site Boundary which typically consider only the worst two hour time period.

The TEDE dose limit for the control room is given as 5.0 rem (TEDE) in 10 CFR 50.67 and GDC-19.

15.4.2.2.4 Methodology

The radiological consequences of the WDG T Rupture are analyzed using the RADTRAD-NAI code and the inputs/assumptions discussed herein. The St. Lucie EPU dose analysis used one bounding analysis, applicable to both St. Lucie Units. Where necessary, analysis inputs which vary between the units were examined, and a bounding value was selected. Atmospheric Dispersion Factors for both unit's release and receptor points were examined and a conservative selection of the more limiting of the choices available was made.

Consistent with Position 1.C of BTP-11-5:

- Only the radioactive noble gases (Xenon and Krypton) are considered.
- A ground level release is assumed.
- No credit for building wake correction factors was assumed.
- A conservative (5%) short term diffusion estimate was applied.
- No deposition was assumed to occur.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 460 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to high radiation signal in the Control Room ventilation system. A 50-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 460 cfm of unfiltered inleakage, and 1760 cfm of filtered recirculation flow.
- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of up to 504 cfm of filtered makeup flow, 460 cfm of unfiltered in leakage, and 1256 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% elemental iodine, and 95% organic iodine. The control room recirculation filtration system will have no effect on the noble gas; therefore, control room recirculation filtration is not modeled.

15.4.2.2.5 Radiological Consequences

The dose locations were specified for the Exclusion Area Boundary (EAB), the Low Population Zone (LPZ), and the Control Room (CR) based upon the most limiting combinations of release points and receptor locations to bound both Unit 1 and Unit 2 analysis. The atmospheric dispersion factors (X/Qs) for the EAB are presented in Table 15.4.2-3. Note that only the 0.0 to 2.0 hour X/Q value is input into RADTRAD-NAI, since this conservatively maximizes the EAB dose for any two hour time period, and has little effect on the 30 day EAB dose for this event.

The X/Q's for the LPZ are presented in Table 15.4.2-4. The breathing rates are from Section 4.1.3 of Reg. Guide 1.183.

Control Room X/Q's are presented in Table 15.4.2-5 and 15.4.2-6. Control Room breathing rates and occupancy factor are from Regulatory Guide 1.183 Section 4.2.6.

Table 15.2.4-7 presents the results for the WGDT rupture. The control room shine dose included in the summary table conservatively reflects the contribution from the external cloud for the LOCA event. As shown in this table, the results for EAB dose, LPZ dose, and Control Room dose are all well within the appropriate regulatory guidance.

An additional output of this calculation is the determination of a candidate Technical Specification limit on Waste Gas Decay Tank inventory of each tank, as described in BTP-11-5 Rev 3 (March 2007), Position 2.A.iii. The analysis for St. Lucie Units 1 and 2 EPU conditions defines a limit on Waste Gas Decay Tank Inventory of 165,000 Curies (Dose Equivalent Xe-133, as defined by TSTF-490).

The calculation of record for this analysis is in Section 15.5, Reference (8).

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TABLE 15.4.2-1
WGDT RUPTURE – INPUTS AND ASSUMPTIONS

INPUT/ASSUMPTION	VALUE
Core Power Level	3030 MW _{th} (3020 + 0.3%)
Core Average Fuel Burnup	49,000 MDW/MTU
Fuel Enrichment	1.5 – 5.0 w/o
WGDT activity	Table 15.4.2-2
Atmospheric Dispersion Factors	
Offsite	Table 15.4.2-3 and 15.4.2-4
Onsite	Table 15.4.2-5 and 15.4.2-6
Control Room Ventilation System	
Time of Control Room Ventilation System Isolation	50 seconds
Time of Control Room Filtered Makeup Flow	1.5
Control Room Unfiltered Inleakage	460 cfm
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

TABLE 15.4.2-2

WGDT SOURCE TERM – 165,000 CURIES Xe-133 EQUIVALENT

ISOPTOPE	RCS/WGDT EVENT INVENTORY		TRM MAXIMUM INVENTORY
	(Ci)		(Ci)
Kr-85	41.60		
Kr-85m	1.268		
Kr-87	0.7574		
Kr-88	2.247		
Xe-131m	3.022		
Xe-133	238.1		165,000
Xe-133m	3.479		
Xe-135	9.235		
Xe-135m	0.8553		
Xe-138	0.5168		
NOMINAL EQ. Xe-133	90,921.0	Analysis Basis (TRM) EQ. Xe-133	165,000
ROUNDED DOWN EQ. Xe-133	90,900		

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TABLE 15.4.2-3

EAB X/Q

TIME (HOURS)	ST. LUCIE COMMON EAB X/Q (SEC/M³)
0.0	1.05E-04
720.0	1.05E-04

TABLE 15.4.2-4

LPZ X/Q

TIME (HOURS)	ST. LUCIE COMMON LPZ X/Q (SEC/M³)
0.0	1.01E-04
2.0	5.74E-05
8.0	4.32E-05
24.0	2.33E-05
96.0	9.62E-06
720.0	9.62E-06

TABLE 15.4.2-5

COMMON ST. LUCIE HVAC E/Q TABLE

TIME (HOURS)	ST. LUCIE COMMON HVAC X/Q (SEC/M³)
0	3.770E-03
0.01389	3.770E-03
1.5	3.770E-03
2	3.195E-03
8	1.390E-03
24	1.101E-03
96	8.870E-04
720	8.870E-04

**TABLE 15.4.2-6
COMMON ST. LUCIE CONTROL ROOM UNFILTERED INLEAKAGE X/Q**

TIME (HOURS)	UNFILTERED INLEAKAGE X/Q (SEC/M³)
0	3.61E-03
2	2.92E-03
8	1.23E-03
24	9.38E-04
96	7.66E-04
720	7.66E-04

**TABLE 15.4.2-7
ST. LUCIE UNITS 1 AND 2 WASTE GAS DECAY TANK FAILURE**

DOSE CONTRIBUTIONS	TEDE DOSE (REM)		
	EAB 30 DAYS	LPZ 30 DAYS	CR 30 DAYS
WDGT FAILURE	0.055086	0.052967	0.19085
CONTROL ROOM CLOUD SHINE	N/A	N/A	0.078
TOTAL	0.055	0.053	0.269
ACCEPTANCE CRITERIA	0.1*	0.1*	5.0**
CONTROL ROOM UNFILTERED INLEAKAGE = 460 CFM			

* The 0.1 REM (TEDE) 30 day dose limit is specified in NUREG-0800, BTP-11-5 Rev 3 (March 2007) Position B.1.A for the EAB. The LPZ limit is assumed in this calculation to be the same value.

** The 5.0 REM (TEDE) CR limit is not specified in either the BTP or Reg Guide 1.183 for this event, but is the specified Control Room limit for all other AST events.

EC300419

15.4.3 FUEL HANDLING ACCIDENT

15.4.3.1 Identification of Causes

The likelihood of a fuel handling accident is minimized by administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures under direct surveillance of a qualified supervisor. Before any refueling operations begin, verification of complete CEA insertion is obtained by ensuring all CEAs are tripped and their rod bottom lights are lit. Boron concentration in the coolant is raised to the refueling concentration and is verified by chemical analysis. At the refueling boron concentration, the core would be subcritical even with all CEA's withdrawn.

After the vessel head is removed, the CEA drive shafts are removed from their respective assemblies. A load cell is used to indicate that the drive shaft is free of the CEA as the lifting force is applied.

The maximum elevation to which the fuel assemblies can be raised is limited by the design of the fuel handling hoists and manipulators to assure that the minimum depth of water above the top of a fuel assembly required for shielding is always present (see Section 9.1.4). This constraint is present in the fuel handling areas both inside containment and in the fuel handling building. Supplementing the physical limits on fuel withdrawal, radiation monitors located at the fuel handling areas provide both audible and visual warning of high radiation levels in the event of a low water level in the refueling cavity and fuel pool. Fuel pool structural integrity is assured by designing the pool and the spent fuel storage racks as Class 1 structures.

The design of the spent fuel storage racks and handling facilities in both the containment and fuel handling building is such that subcriticality would be maintained if the pool were flooded with unborated water. Natural convection of the surrounding water provides adequate cooling of fuel during handling and storage. Adequate cooling of the water is provided by forced circulation in the spent fuel pool cooling system. At no time during the transfer from the reactor core to the spent fuel storage rack is there less than 112 inches of water above a fuel assembly.

Fuel failure during refueling as a result of inadvertent criticality or overheating is not possible. The possibility of damage to a fuel assembly as a consequence of mishandling is minimized by thorough training, detailed procedures and equipment design. The design precludes the handling of heavy objects such as shipping casks over the spent fuel pool storage racks. Administrative controls prevent the movement of heavy loads over the cask pit whenever the cask pit rack is installed in the cask area of the SFP. Inadvertent disengagement of a fuel assembly from the fuel handling machine is prevented by mechanical interlocks, consequently, the possibility of dropping and damaging of a fuel assembly is remote.

Should a spent fuel assembly be damaged during handling, radioactive release could occur in either the containment or the fuel handling building. The ventilation exhaust air from both of these areas is monitored before release to the atmosphere (Section 11.4). Area radiation monitors provide alarm and indication of increased activity level. The affected area would then be evacuated.

The original fuel handling accident analyzed the off-site dose consequences from the event occurring in the fuel handling building. A second fuel handling building analysis was performed for an extended burn-up source term. The current analysis of the Fuel Handling Accident was performed to support a Technical Specification Amendment to allow the containment personnel airlock doors to remain open during refueling operations and core alterations. This analysis evaluates the off-site dose from a fuel handling accident in the refueling canal with a water level of 23 feet.

The previous analysis of a dropped fuel assembly in the fuel handling building assumed that a fuel assembly was dropped into the spent fuel pool during fuel handling. The analysis evaluated the damage potential to the dropped fuel assembly and also to the stored fuel assemblies. Interlocks and procedural and administrative controls make such an event highly unlikely; however, if an assembly were damaged to the extent that one or more fuel rods were broken, the accumulated fission gases and iodines in the gap would be released to the surrounding water. Release of the solid fission products in the fuel would be negligible since the low fuel temperature during refueling greatly limits their diffusion.

The fuel assemblies are stored in the spent fuel racks at the bottom of the fuel pool. The top of the rack extends above the tops of the stored fuel assemblies, so that a dropped fuel assembly could not strike more than one fuel assembly in the storage rack. In this case, impact could occur only between the ends of the fuel assemblies, the bottom end fitting of the dropped fuel assembly striking the top end fitting of the stored fuel assembly. The results of an analysis of the "end-on" energy absorption capability of a fuel assembly have shown that a fuel assembly is capable of absorbing the kinetic energy of the drop and that there will be no fuel damage. The worst fuel handling accident that could occur in the spent fuel pool is the dropping of a fuel assembly to the fuel pool floor. After striking the pool floor vertically, the assembly would rotate into a horizontal attitude; during this rotation, it is postulated that the assembly strikes a protruding structure. The fuel storage pool is designed with no protruding structures and, hence, the shape and nature of the assumed protruding structure is indeterminate. For this analysis, therefore, a line load was assumed.

To obtain an estimate of the number of fuel rods which might fail in the event a fuel assembly were dropped, the energy required to crush a fuel rod and bend the entire assembly has been determined. The point of impact was assumed at the most effective location for fuel rod damage, i.e., the center of percussion. Resistance to crushing offered by the fuel pellet is considered in the analysis. The model results in a conservative upper limit for the number of fuel rod failures. Since it is not possible to apply a line load beyond the outer row of fuel rods, failure by crushing cannot be experienced beyond the outer row. The failure mode of rods in other than the outer rows will be by bending rather than crushing.

Approximately 36,000 in-lb of kinetic energy from rotation must be absorbed. The energy required to bend the assembly and crush the outer row of fuel rods to failure is 4,600 in-lb. Failure of the second row of fuel rods by bending alone requires more than 70,000 in-lb. Thus, for a fuel assembly dropped into the spent fuel pool, no more than 14 fuel rods, i.e., one outer row of rods, would be expected to fail.

15.4.3.2 Radiological Analysis

15.4.3.2.1 Background

This event consists of the drop of a single fuel assembly either in the Fuel Handling Building (FHB) or inside of Containment. This analysis considers both a dropped fuel assembly inside the containment with the maintenance hatch open, and an assembly drop inside the FHB without credit for filtration of the Fuel Handling Building exhaust. The source term released from the overlying water pool is the same for both the FHB and the containment cases. RG 1.183 guidance provides the same 2-hour criteria for the direct unfiltered release of the activity to the environment for either location.

A minimum water level of 23 feet is maintained above the damaged fuel assembly for both the containment and FHB release locations. This water level ensures an elemental iodine decontamination factor of 285 per the guidance provided in NRC Regulatory Issue Summary 2006-04 (Reference 106). The St. Lucie Unit 1 AST dose analysis methodology is presented in Reference 107.

15.4.3.2.2 Compliance with RG 1.183 Regulatory Positions

The FHA dose consequence analysis is consistent with the guidance provided in RG 1.183 Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident," as discussed below:

1. Regulatory Position 1.1 - The amount of fuel damage is assumed to be all of the fuel rods in a single fuel assembly.
2. Regulatory Position 1.2 - The fission product release from the breached fuel is based on Regulatory Positions 3.1 and 3.2 of RG 1.183. The gap activity available for release is specified by Table 3 of RG 1.183. Gap release fractions are doubled to account for high burnup fuel rods. This activity is assumed to be released instantaneously.
3. Regulatory Position 1.3 - The chemical form of radioiodine released from the damaged fuel into the spent fuel pool is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. The cesium iodide is assumed to completely dissociate in the spent fuel pool resulting in a final iodine distribution of 99.85% elemental iodine and 0.15% organic iodine.
4. Regulatory Position 2 - A minimum water depth of 23 feet is maintained above the damaged fuel assembly. Therefore, a decontamination factor of 285 is applied to the elemental iodine and a decontamination factor of 1 is applied to the organic iodine. As a result, the breakdown of the iodine species above the surface of the water is 57% elemental and 43% organic. Guidance for the use of 285 for the elemental iodine decontamination factor is provided in NRC Regulatory Issue Summary 2006-04, "Experience with Implementation of Alternate Source Terms."
5. Regulatory Position 3 - All of the noble gas released is assumed to exit the pool without mitigation. All of the non-iodine particulate nuclides are assumed to be retained by the pool water.

6. Regulatory Position 4.1 - The analysis models the release to the environment over a 2-hour period.
7. Regulatory Position 4.2 - No credit is taken for filtration of the release.
8. Regulatory Position 4.3 - No credit is taken for dilution of the release.
9. Regulatory Position 5.1 - The containment maintenance hatch is assumed to be open at the time of the fuel handling accident.
10. Regulatory Position 5.2 - No automatic isolation of the containment is assumed for the FHA.
11. Regulatory Position 5.3 - The release from the fuel pool is assumed to leak to the environment over a two-hour period.
12. Regulatory Position 5.4 - No ESF filtration of the containment release is credited.
13. Regulatory Position 5.5 - No credit is taken for dilution or mixing in the containment atmosphere.

15.4.3.2.3 Methodology

The input assumptions used in the dose consequence analysis of the FHA are provided in Table 15.4.3-1. It is assumed that the fuel handling accident occurs at 72 hours after shutdown of the reactor. 100% of the gap activity specified in Table 3 of RG 1.183 is assumed to be instantaneously released from a single fuel assembly into the fuel pool. A minimum water level of 23 feet is maintained above the damaged fuel for the duration of the event. 100% of the noble gas released from the damaged fuel assembly is assumed to escape from the pool. All of the non-iodine particulates released from the damaged fuel assembly are assumed to be retained by the pool. Iodine released from the damaged fuel assembly is assumed to be composed of 99.85% elemental and 0.15% organic. All activity released from the pool is assumed to leak to the environment over a two-hour period. No credit for dilution in the containment or FHB is taken.

The FHA source term listed in Table 15.4.3-2 meets the requirements of Regulatory Position 1 of Appendix B to RG 1.183. The analysis includes a decay time of 72 hours before the beginning of fuel movement. Since the FHA source term presented in Table 15.4.3-2 does not include this decay time, it is accounted for in the RADTRAD-NAI model.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 460 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 50-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 460 cfm of unfiltered inleakage, and 1760 cfm of filtered recirculation flow.
- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of up to 504 cfm of filtered makeup flow, 460 cfm of unfiltered inleakage, and 1256 cfm of filtered recirculation flow.

- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% elemental iodine, and 95% organic iodine.

15.4.3.2.4 Radiological Consequences

The Control Room atmospheric dispersion factors (χ/Q_s) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake.

When the Control Room Ventilation System is in normal mode, the most limiting χ/Q corresponds to the worst air intake to the control room. When the ventilation system is isolated, the limiting χ/Q corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. Development of control room atmospheric dispersion factors is discussed in Appendix 2J. The χ/Q for the FHA events are summarized in Table 15.4.3-3.

For the EAB dose analysis, the χ/Q factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period. The LPZ dose is determined using the χ/Q factors for the appropriate time intervals. These χ/Q factors are provided in Appendix 2I.

The radiological consequences of the FHA are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 15.4.3-4, the results for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

The calculation of record for this analysis is in Section 15.5, Reference (8).

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15.4.3.3 Effect of Replacement Steam Generators

The dose consequences of the fuel handling accident are a function of the fuel rod gap fission gas activity of the hottest fuel assembly and of the refueling pool decontamination factor. The replacement steam generators do not affect fission gas activity or the refueling pool decontamination factor. Therefore, this accident is not affected by the replacement steam generators.

15.4.3.4 Effect of Dry Storage of Irradiated Fuel

Any irradiated nuclear fuel that is to be placed in a TN 32PTH dry shielded canister (DSC) must first cool in the spent fuel storage racks for at least five years. Fission product decay during this extended cooling interval ensures that dose consequences resulting from the postulated cask drop accident in the Transnuclear Final Safety Analysis Report will meet the plant site boundary dose limits. For the HI-STORM system, a Cask Drop is not credible as it is lifted by the single-failure proof cask handling crane in conjunction with Lifting Devices meeting ANSI N14.6 and NUREG0612 requirements.

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15.4.3.5 References for Section 15.4.3.4

1. NUHOMS® HD System Generic Technical Specifications (included as Appendix A to Certificate No. 1030 - Certificate of Compliance for Spent Fuel Storage Casks, dated 01/10/2007)

**Table 15.4.3-1
Fuel Handling Accident (FHA) – Inputs and Assumptions**

Input/Assumption	Value
Core Power Level Before Shutdown	3030 MW _{th} (3020 + 0.3%)
Core Average Fuel Burnup	49,000 MWD/MTU
Discharged Fuel Assembly Burnup	45,000 – 62,000 MWD/MTU
Fuel Enrichment	1.5 – 5.0 w/o
Maximum Radial Peaking Factor	1.65
Number of Fuel Assemblies in the Core	217
Number of Fuel Assemblies Damaged	1
Delay Before Spent Fuel Movement	72 hours
FHA Source Term for a Single Assembly	Table 15.4.3-2
Water Level Above Damaged Fuel Assembly	23 feet minimum
Iodine Decontamination Factors	Elemental – 285 Organic – 1
Noble Gas Decontamination Factor	1
Chemical Form of Iodine In Pool	Elemental – 99.85% Organic – 0.15%
Chemical Form of Iodine Above Pool	Elemental – 57% Organic – 43%
Atmospheric Dispersion Factors Offsite Onsite	Appendix 21 Table 15.4.3-3
Control Room Ventilation System Time of Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	50 seconds 1.5 hours 460 cfm
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 15.4.3-2

Fuel Handling Accident Source Term

Nuclide	Activities (Curies)	Nuclide	Activities (Curies)	Nuclide	Activities (Curies)
Kr-85	9.417E+03	Xe-135	3.341E+05	Te-133	7.486E+05
Kr-85m	1.508E+05	Cs-134	2.536E+05	Te-134	9.973E+05
Kr-87	2.864E+05	Cs-136	6.227E+04	Te-125m	3.911E+03
Kr-88	4.026E+05	Cs-137	1.052E+05	Te-133m	4.424E+05
Rb-86	2.142E+03	Ba-139	1.094E+06	Ba-141	9.918E+05
Sr-89	5.521E+05	Ba-140	1.054E+06	Ba-137m	9.973E+04
Sr-90	7.554E+04	La-140	1.101E+06	Pd-109	4.216E+05
Sr-91	6.856E+05	La-141	9.966E+05	Rh-106	7.574E+05
Sr-92	7.494E+05	La-142	9.600E+05	Rh-103m	1.081E+06
Y-90	7.880E+04	Ce-141	1.014E+06	Tc-101	1.117E+06
Y-91	7.212E+05	Ce-143	9.179E+05	Eu-154	1.586E+04
Y-92	7.531E+05	Ce-144	8.526E+05	Eu-155	1.100E+04
Y-93	8.804E+05	Pr-143	9.128E+05	Eu-156	3.622E+05
Zr-95	1.016E+06	Nd-147	4.023E+05	La-143	9.110E+05
Zr-97	1.011E+06	Np-239	1.851E+07	Nb-97	1.021E+06
Nb-95	1028E+06	Pu-238	4.719E+03	Nb-95m	7.268E+03
Mo-99	1.202E+06	Pu-239	2.911E+02	Pm-147	9.215E+04
Tc-99m	1.052E+06	Pu-240	5.480E+02	Pm-148	1.879E+05
Ru-103	1.200E+06	Pu-241	1.357E+05	Pm-149	4.224E+05
Ru-105	9.714E+05	Am-241	1.532E+02	Pm-151	1.544E+05
Ru-106	6.909E+05	Cm-242	6.798E+04	Pm-148m	2.259E+04
Rh-105	8.747E+05	Cm-244	2.488E+04	Pr-144	8.583E+05
Sb-127	8.842E+04	I-130	5.275E+04	Pr-144m	1.024E+04
Sb-129	2.399E+05	Kr-83m	7.273E+04	Sm-153	5.158E+05
Te-127	8.796E+04	Xe-138	1.004E+06	Y-94	8.931E+05
Te-127m	1.200E+04	Xe-131m	7.470E+03	Y-95	9.671E+05
Te-129	2.361E+05	Xe-133m	4.074E+04	Y-91m	3.980E+05
Te-129m	3.503E+04	Xe-135m	2.671E+05	Br-82	5.881E+03
Te-131m	1.012E+05	CS-138	1.117E+06	Br-83	7.247E+04
Te-132	9.227E+05	Cs-134m	5.683E+04	Br-84	1.241E+05
I-131	6.654E+05	Rb-88	4.100E+05	Am-242	9.393E+04
I-132	9.425E+05	Rb-89	5.234E+05	Np-238	5.019E+05
I-133	1.255E+06	Sb-124	2.681E+03	Pu-243	8.717E+05
I-134	1.358E+06	Sb-125	1.767E+04		
I-135	1.183E+06	Sb-126	1.359E+03		
Xe-133	1.260E+06	Te-131	5.853E+05		

Table 15.4.3-3
 χ/Qs for Containment Release

Time (hours)	χ/Q (sec/m³)
0.0	1.90E-3
72.013889	1.21E-3
73.5	8.22E-4
74.0	6.57E-4
80.0	2.87E-4
96.0	1.92E-4
168.0	1.74E-4
792.0	1.74E-4

Control Room χ/Qs for FHB Release

Time (hours)	χ/Q (sec/m³)
0.0	4.99E-3
72.013889	3.27E-3
73.5	2.01E-3
74.0	1.44E-3
80.0	6.25E-4
96.0	4.34E-4
168.0	3.33E-4
792.0	3.33E-4

Table 15.4.3-4
Fuel Handling Accident Dose Consequences

Cass	EAB Dose⁽¹⁾ (REM TEDE)	LPZ Dose⁽²⁾ (REM TEDE)	Control Room Dose⁽²⁾ (REM TEDE)
FHA Containment Release	0.56	0.58	1.43
FHA Fuel Handling Building Release	0.56	0.55	3.47
Acceptance Criteria	6.3	6.3	5

⁽¹⁾ Worst 2-hour dose

⁽²⁾ Integrated 30-day dose

15.4.4 STEAM GENERATOR TUBE FAILURE

15.4.4.1 Identification of Causes

The steam generator tube failure is a penetration of the barrier between the reactor coolant system and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking steam generator tube allows the transfer of reactor coolant into the main steam system. Radioactivity contained in the reactor coolant mixes with water in the shell side of the affected steam generator. This radioactivity is transported by steam to the turbine and then to the condenser, or directly to the condenser via the main steam dump and bypass system. Noncondensable radioactive gases in the condenser are removed by the condenser air ejector discharge to the plant vent.

Detection of reactor coolant leakage to the steam system is facilitated by radiation monitors in the steam generator blowdown lines (see Section 10.4.7), in the condenser air ejector discharge lines (see Section 10.4.2 and 10.4.3) and in the main steam line radiation monitors. These monitors initiate alarms in the control room and alert the operator of abnormal activity levels and that corrective action is required.

The behavior of the reactor coolant varies depending upon the size of the rupture. For leak rates up to the capacity of the charging pumps in the chemical and volume control system, reactor coolant inventory can be maintained and an automatic reactor trip will not occur. The gaseous fission products would be released to atmosphere from the main steam system via the condenser air ejector discharge to the plant vent. Those fission products not discharged in this way would be retained by the main steam, feedwater and condensate systems.

For leaks that exceed the capacity of the charging pumps, pressurizer water level and pressurizer pressure decrease and an automatic reactor trip results. The turbine then trips and the main steam dump and bypass valves open, discharging steam directly into the condenser. In addition to the radiation monitors, the steam generator water level indicators aid in the detection of these larger leaks since the water inventory in the leaking steam generator will increase more rapidly than that of the intact steam generator following the reactor trip. (Prior to reactor trip the water level in each steam generator is automatically maintained at a constant level.)

As the break flow begins to depressurize the RCS, the charging pumps activate in order to make-up the lost inventory. If the RCS inventory and pressure are stabilized via the charging pumps, no reactor trip will occur. However, if the break flow exceeds the capacity of the charging pumps, the RCS pressure and inventory will continue to decrease resulting in a reactor trip on a low RCS pressure signal (TM/LP). At normal operating conditions, the leak rate through the double-ended rupture of one tube is greater than the maximum flow available from the three charging pumps.

Following the reactor trip, the turbine will trip and, in the case where offsite power is lost, the reactor coolant pumps will coast down and make-up flow will terminate until emergency diesel generator power is available. If offsite power is available, a fast transfer to the offsite power will keep the reactor coolant pumps running and the makeup flow available.

The loss of offsite power results in the loss of condenser vacuum and the steam dump to condenser valves are closed to protect the condenser. The continued mass and energy transfer between the RCS and secondary side results in an increase in the affected SG pressure and discharge to the atmosphere via the MSSVs and ADVs.

The reactor trip produces a further decrease in reactor coolant system pressure. As pressurizer water level drops, the pressurizer heaters are uncovered and automatically deenergized, and the reactor coolant system pressure begins to drop more rapidly. As the RCS pressure continues to decrease, a low pressurizer pressure signal activates the SIS. The emergency diesels start and HPSI flow begins once the shutoff head of the HPSI pumps has been reached. In accordance with the EOPs, the operators will isolate the affected steam generator by closing the associated MSIV, and will take a series of actions to regain control of the plant systems and to bring the RCS to a condition allowing for initiation of the RHR system.

15.4.4.2 Analysis of Effects and Consequences

A steam generator tube rupture overflow analysis and steam generator tube rupture mass release analysis are performed.

15.4.4.2.1 Steam Generator Tube Rupture

The event is postulated to be a double-ended rupture of one steam generator tube at full power. The event is characterized by a depressurization of the RCS with reactor trip on a TM/LP signal. Loss of offsite power occurs at reactor trip, after which the SBCS is not credited and steam release is via the steam generators MSSVs. Operators isolate the affected steam generator by closing the associated MSIV, and begin cooldown using the ADV on the unaffected steam generator.

Detailed analyses were performed with the approved methodology using the S-RELAP5 code (Reference 117). The S-RELAP5 code was used to model the key primary and secondary system components, RPS and ESF actuation trips and core kinetics. Calculations were performed to determine the steam releases from event initiation to two hours. Separate calculations were performed to determine the steam releases resulting from the cooldown of the plant to an RCS temperature of 212°F.

The input parameters and biasing were consistent with the approved methodology.

- Initial Conditions - This event was analyzed from HFP to produce the highest decay heat level and the most significant atmospheric steam release.
- Reactivity Feedback - The reactivity feedback coefficients were biased according to the approved methodology. BOC moderator density feedback was assumed for this event, although the reactivity feedback is not a significant parameter.
- Reactor Protection System Trips and Delays - This event is primarily protected by the TM/LP RPS trip.
- Decay Heat - Decay heat was calculated using the 1973 ANS standard plus actinides in accordance with the approved methodology.
- Break Location and Characteristics - Two potential break locations were analyzed. The first location assumed the break occurred on the upside of the steam generator tube bundle at the exit of the tubesheet (i.e., hot-side break). The second location assumed the break occurred on the downside of the steam generator tube bundle at the entrance to the tubesheet (i.e., cold-side break). The double-ended break of a single steam generator tube was modeled such that RCS liquid was lost from both the upstream and downstream sides of the break. Moody critical flow was assumed at the rupture tube junction.
- Offsite Power - Offsite power was assumed to be lost at reactor trip resulting in a loss of condenser vacuum. Loss of condenser vacuum results in the loss of the SBCS for removal of decay heat. Heat removal from the RCS is achieved by action of the steam generator MSSVs up to the time of operator action at which time the ADV in the unaffected steam generator was credited.

- Operator Actions - An operator action time of up to 45 minutes was analyzed. Operators were assumed to isolate the affected steam generator by closing the associated MSIV, and to begin cooldown using the ADV on the unaffected steam generator.

This event is classified as a Postulated Accident. The principally challenged acceptance criterion for this event is with respect to radiological consequences. This event is protected by the TM/LP trip and does not represent a significant challenge to the SAFDLs, so no fuel failure is expected.

Figures 15.4.4-1 through 15.4.4-11 present the transient behavior of reactor power, pressurizer liquid level, pressurizer pressure, RCS loop temperatures, RCS total loop flow rate, steam generator pressures, reactivity feedback, steam generator masses, MSSV flow rate, total break flow rate, and integrated break flow.

The calculations of record for these analyses are listed in Section 15.5, Reference (8).

EC300419

15.4.4.2.2 Steam Generator Tube Rupture Overfill

A conservative SGTR overfill analysis was performed with the approved methodology using the S-RELAP5 code (Reference 117).

A single case was analyzed. Parameter biasing, assumptions, and an assumed single failure were designed to produce a conservatively high break flow rate, maximize Auxiliary Feedwater (AFW) flow to the ruptured SG, and minimize the Margin to Overfill (MTO) at the time operators terminate AFW flow to the ruptured SG. Assumptions regarding operator actions and mitigating systems and functions, along with a limiting single failure, produce the most challenging scenario regarding SG overfill. The case analyzed is described below.

The SGTR event is initiated by a double-ended break of a single steam generator tube (shortest tube) on the top side of the tubesheet. The break is assumed to be at the coldside of the U-tube at the top surface of the tube sheet above the SG outlet plenum. A cold-side break is analyzed because it produces a higher total break flow rate than a hot-side break, which is in the conservative direction for the overfill analysis.

Loss of Offsite Power (LOOP) is assumed at reactor trip in this analysis. The assumption of LOOP at reactor trip is conservative relative to offsite power being available, as safety grade overfill protection for Main Feedwater (MFW) would be available and would prevent SG overfill in a no-LOOP case.

There are no operator actions or mitigating systems or functions simulated directly in this analysis to cooldown and depressurize the Reactor Coolant System (RCS) to equilibrate RCS and ruptured SG pressures and terminate break flow. Therefore, the calculated integrated break flow will bound an actual integrated break flow (out to the time that operators terminate AFW flow to the ruptured SG) that would occur when operators take mitigating steps in accordance with the Emergency Operating Procedures (EOPs). The only operator action directly accounted for in the analysis is termination of Turbine-Driven (TD) AFW flow to the ruptured SG following Auxiliary Feedwater Actuation Setpoint (AFAS) reset when the ruptured SG Narrow Range (NR) level reaches 35%, plus a 15 minute delay time for operator action.

Along with no operator actions or mitigating systems or functions being directly simulated to terminate break flow, the most challenging single failure to overfill of the ruptured SG is a failed open TD AFW flow control valve on AFAS reset. This produces the largest AFW flow to the ruptured SG, which reduces the ruptured SG pressure and tends to increase the break flow rate. Without this single failure, the AFAS reset logic will prevent AFW flow to the ruptured SG when the nominal SG level is above 29% NR (analysis value 35% NR).

The maximum ruptured SG liquid volume at the time operators terminate TD AFW flow to the ruptured SG was calculated to be 5598.2 ft³. The total secondary side volume of the SG is 7733.7 ft³. Thus, the MTO is 2135.5 ft³ at the time of ~28 minutes when operators terminate AFW flow to the ruptured SG. This MTO of greater than 2100 ft³ calculated with conservative break flow, would be sufficient to prevent SG overflow considering the fact that operators will be taking action to reduce the SG level and reduce the pressure difference between the RCS and the ruptured SG, which would result in reduced break flow. The conservative break flow at the time of termination of AFW flow was ~42 lbm/sec. Even assuming continuation of this break flow up to 45 minutes into the transient with no other operator action, the MTO will remain greater than 1000 ft³.

The sequence of events for the SGTR overflow analysis is shown in Table 15.4.4-2b.

Figure 15.4.4-12 shows the liquid volume in the ruptured SG versus time relative to the total geometric volume of the SG. The MTO is 2135.5 ft³ at the time the operators terminate AFW flow to the ruptured SG (27.6 minutes).

The calculation of record for this analysis is in Section 15.5, Reference (8).

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15.4.4.5 Radiological Analysis

15.4.4.5.1 Background

This event is assumed to be caused by the instantaneous rupture of a Steam Generator tube that relieves to the lower pressure secondary system. No melt or clad breach is postulated for the St. Lucie Unit 1 SGTR event. The St. Lucie Unit 1 AST dose analysis methodology is presented in Reference 107.

15.4.4.5.2 Compliance with RG 1.183 Regulatory Positions

The SGTR dose consequence analysis followed the guidance provided in RG 1.183, Appendix F, "Assumptions for Evaluating the Radiological Consequences of a PWR Steam Generator Tube Rupture Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1. No fuel damage is postulated to occur for the St. Lucie Unit 1 SGTR event.
2. Regulatory Position 2 - No fuel damage is postulated to occur for the St. Lucie Unit 1 SGTR event. Two cases of iodine spiking are assumed.
3. Regulatory Position 2.1 - One case assumes a reactor transient prior to the postulated SGTR that raises the primary coolant iodine concentration to the maximum allowed by Technical Specification 3.4.16 value of 60.0 $\mu\text{Ci/gm}$ DE I-131. This is the pre-accident spike case. EC
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478
4. Regulatory Position 2.2 - One case assumes the transient associated with the SGTR causes an iodine spike. The spiking model assumes the primary coolant activity is initially at the Technical Specification 3.4.16 value of 1.0 $\mu\text{Ci/gm}$ DE I-131. Iodine is assumed to be released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. This is the accident-induced spike case. EC
299
478
5. Regulatory Position 3 - The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
6. Regulatory Position 4 - Iodine releases from the steam generators to the environment are assumed to be 97% elemental and 3% organic.
7. Regulatory Position 5.1 - The primary-to-secondary leak rate is apportioned between the SGs as specified by Technical Specification 5.5.6 (0.5 gpm total, 0.25 to any one SG). Thus, the tube leakage is apportioned equally between the two SGs. EC
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8. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is based upon RCS conditions, consistent with the plant design basis.
9. Regulatory Position 5.3 - The primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F. An input parameter for the termination of the affected SG activity release states that the affected SG is isolated within 45 minutes by operator action. This isolation terminates releases from the affected SG, while primary-to-secondary leakage continues to provide activity for release from the unaffected SG.
10. Regulatory Position 5.4 - The release of fission products from the secondary system is evaluated with the assumption of a coincident loss-of-offsite power (LOOP).

11. Regulatory Position 5.5 - All noble gases released from the primary system are assumed to be released to the environment without reduction or mitigation.
12. Regulatory Position 5.6 - Regulatory Position 5.6 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows:
 - Appendix E, Regulatory Position 5.5.1 - A portion of the primary-to-secondary ruptured tube flow through the SGTR is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary. For the unaffected steam generator used for plant cooldown, flashing is considered immediately following plant trip when tube uncover is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.
 - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG, enter the steam space, and be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited.
 - Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage and ruptured tube flow that does not flash is assumed to mix with the bulk water.
 - Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SGs. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.
 - Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncover in the intact SG is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 1. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water of the SG into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator. A conservative uncover time of 60 minutes was assumed in the analysis.

15.4.4.5.3 Other Assumptions

1. This evaluation assumes that the RCS mass remains constant throughout the event.
2. For the purposes of determining the iodine concentrations, the SG mass is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore water level above the top of the tubes in the unaffected steam generator within a conservative time of one hour following a reactor trip.
3. Data used to calculate the iodine equilibrium appearance rates are provided in Table 15.4.4-5, "Iodine Equilibrium Appearance Assumptions." The iodine spike activity appearance rates are provided in Table 15.4.4-6.

15.4.4.5.4 Methodology

Input assumptions used in the dose consequence analysis of the SGTR event are provided in Table 15.4.4-3. This event is assumed to be caused by the instantaneous rupture of a steam generator tube releasing primary coolant to the lower pressure secondary system. In the unlikely event of a concurrent loss of power, the loss of circulating water through the condenser would eventually result in the loss of condenser vacuum, thereby causing steam relief directly to the atmosphere from the ADVs. This direct steam relief continues until the faulted steam generator is isolated at 45 minutes.

A thermal-hydraulic analysis is performed to determine a conservative maximum break flow, break flashing flow, and steam release inventory through the faulted SG relief valves. Additional activity, based on the proposed primary-to-secondary leakage limits, is released via steaming from the ADVs until the RCS is cooled to 212°F.

Per UFSAR, Section 15.4.4.6, no fuel failure is postulated for the SGTR event. Consistent with RG 1.183 Appendix F, Regulatory Position 2, if no, or minimal, fuel damage is postulated for the limiting event, the activity release is assumed as the maximum allowed by Technical Specifications for two cases of iodine spiking: (1) maximum pre-accident iodine spike, and (2) maximum accident-induced, or concurrent, iodine spike.

For the case of a pre-accident iodine spike, a reactor transient is assumed to have occurred prior to the postulated SGTR event. The primary coolant iodine concentration is increased to the maximum value of 60 $\mu\text{Ci/gm}$ DE I-131 permitted by Technical Specification 3.4.16 (see Table 15.4.4-7). Primary coolant is released into the ruptured SG by the tube rupture and by a fraction of the total proposed allowable primary-to-secondary leakage. Activity is released to the environment from the ruptured SG via direct flashing of a fraction of the released primary coolant from the tube rupture and also via steaming from the ruptured SG ADVs until the ruptured steam generator is isolated at 45 minutes. The unaffected SG is used to cool down the plant during the SGTR event. Primary-to-secondary tube leakage is also postulated into the intact SG. Activity is released via steaming from the unaffected SG ADVs until the RCS is cooled below 212°F. These release assumptions are consistent with the requirements of RG 1.183.

For the case of the accident-induced spike, the postulated STGR event induces an iodine spike. The RCS activity is initially assumed to be 1.0 $\mu\text{Ci/gm}$ DE I-131 as allowed by Technical Specification 3.4.16. Iodine is released from the fuel into the RCS at a rate of 335 times the iodine equilibrium release rate for a period of 8 hours. Parameters used in the determination of the iodine equilibrium release rate are provided in Table 15.4.4-5. The iodine activities and the appearance rates for the accident-induced (concurrent) iodine spike case are presented in Table 15.4.4-6. All other release assumptions for this case are identical to those for the pre-accident spike case.

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478

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478

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 460 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. For this event, it is conservatively assumed that the CR isolation signal at 522.7 seconds is delayed until the release from the ADVs is initiated. An additional 50-second delay is applied to account for the diesel generator start time, fan start and damper actuation time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 460 cfm of unfiltered inleakage, and 1760 cfm of filtered recirculation flow.
- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of up to 504 cfm of filtered makeup flow, 460 cfm of unfiltered inleakage, and 1256 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% for elemental iodine, and 95% for organic iodine.

15.4.4.5.5 Radiological Consequences

The Control Room atmospheric dispersion factors (χ/Qs) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake.

For the SGTR event, releases are assumed to occur from the condenser prior to reactor trip, and from the ADV that produces the most limiting χ/Qs following reactor trip. When the Control Room Ventilation System is in normal mode, the most limiting χ/Q corresponds to the worst air intake to the control room. When the ventilation system is isolated, the limiting χ/Q corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. Development of control room atmospheric dispersion factors is discussed in Appendix 2J. The χ/Qs for the SGTR releases are summarized in Table 15.4.4-8.

The EAB and LPZ dose consequences are determined using the χ/Q factors provided in Appendix 2I for the appropriate time intervals. For the EAB dose calculation, the χ/Q factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period.

The radiological consequences of the SGTR accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Two activity release cases corresponding to the RCS maximum pre-accident iodine spike and the accident-induced iodine spike, based on Technical Specification 3.4.16 limits, are analyzed. As shown in Table 15.4.4-9, the radiological consequences of the St. Lucie Unit 1 SGTR event for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

The calculation of record for this analysis is in Section 15.5, Reference (8).

EC300419

15.4.4.6 Conclusions

The reactor protective system (i.e., TM/LP trip) intervenes to protect the core from exceeding the DNBR limit. Therefore, fuel failure does not occur in this event. The doses resulting from the activity released as a consequence of a double-ended rupture of one steam generator tube, assuming the maximum allowable Technical Specification activity for the primary concentration at a core power of 3030 Mwt, are significantly below the guidelines of RG 1.183. Thus, the results do not exceed acceptance criteria.

15.4.4.7 DELETED

TABLE 15.4.4-1
KEY PARAMETERS ASSUMED IN THE STEAM GENERATOR TUBE RUPTURE EVENT

PARAMETER	UNITS	VALUE
Core Power	MWt	3029.1
Core Inlet Temperature	°F	551
RCS Flow Rate	gpm	375,000
Pressurizer Pressure	psia	2,225
Pressurizer Level	%	65.6
Doppler Reactivity Coefficient	pcm/°F	-0.8
Moderator Density Reactivity		Based on the most positive Technical Specification MTC

TABLE 15.4.4-2

**SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE EVENT
(45 MIN. OPERATOR ACTION TIME)**

EVENT	HOT SIDE BREAK TIME (SEC)	COLD SIDE BREAK TIME (SEC)
Reactor Operating at HFP conditions	0.0	0.0
Double-ended rupture of SG tube occurred above tubesheet on the hot plenum side		
TM/LP setpoint reached	472.7	387.2
Turbine tripped on TM/LP reactor trip – offsite power is assumed lost	473.6	388.1
RCP's lose power source and coastdown		
MFW lost due to LOOP		
CEA Insertion Begins	474.1	388.6
SG Narrow Range level reached 0.0%	520.0	460.0
AFW flow began	520.0	420.0
Charging flow began	640.0	380.0
HPSI flow began	1180.0	380.0
Operator initiated controlled cooldown @ 100°F/hr <ul style="list-style-type: none"> • MSIV on affected steam generator was closed • ADV on unaffected steam generator began to cycle 	2700.0	2700.0

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TABLE 15.4.4-2a

**KEY PARAMETERS ASSUMED IN THE STEAM GENERATOR TUBE RUPTURE OVERFILL
EVENT**

PARAMETER	UNITS	VALUE
Core Power	MWt	3029.1
Core Inlet Temperature	°F	551
RCS Flow Rate	gpm	375,000
Pressurizer Pressure	psia	2,315
Pressurizer Level	%	68.6
Doppler Reactivity Coefficient	pcm/°F	-0.8
Moderator Density Reactivity		Based on the most positive Technical Specification MTC

TABLE 15.4.4-2b

**SEQUENCE OF EVENTS FOR THE STEAM GENERATOR TUBE RUPTURE OVERFILL EVENT
(45 MIN. OPERATOR ACTION TIME)**

EVENT	TIME (SEC)
Event initiation – Double-ended rupture of SG tube occurs above tubesheet on cold plenum side	0.0
Charging flow begins	0.0
Thermal Margin/Low Pressure (TM/LP) Reactor Protection System (RPS) setpoint reached (including delay time)	229.1
Turbine trips on reactor trip – offsite power is assumed lost	229.1
Reactor Coolant Pumps (RCPs) coastdown due to LOOP	229.1
MFW lost due to LOOP	229.1
Main Steam Safety Valves (MSSVs) open on ruptured SG (SG-1)	232
MSSVs open on intact SG (SG-2)	232
AFAS signal on SG-1 and SG-2	274.2
AFW begins to SG-1 and SG-2	454.2
SG-1 NR level reaches 35% following reactor trip	757.3
Safety Injection Actuation Signal (SIAS)	784.8
Pressurizer empties	806
High Pressure Safety Injection (HPSI) flow begins	1088
TD AFW to SG-1 is terminated	1657.3

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Table 15.4.4-3

Steam Generator Tube Rupture (SGTR) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3030 MW _{th} (3020 + 0.3%)
Initial RCS Equilibrium Activity	1.0 μCi/gm DE I-131 and 518.9 μCi/gm DE Xe-133 gross activity (Table 15.4.1-9)
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131 (Table 15.4.6-8)
Maximum Pre-Accident Spike Iodine Concentration	60μCi/gm DE I-131
Maximum Equilibrium Iodine Concentration	1.0μCi/gm DE I-131
Iodine Spike Appearance Rate	335 times
Duration of Accident-Initiated Spike	8 hours
Integrated Break Flow and Steam Release	Table 15.4.4-4
Break Flow Flashing Fraction	Prior to Reactor Trip - 17% Following Reactor Trip – 6%
Time to Terminate Break Flow	45 minutes
Steam Generator Tube Leakage Rate	0.25 gpm per SG
Time to Terminate Tube Leakage	12.4 hours
Time to Re-cover Intact SG Tubes	1 hour
Steam Generator Secondary Side Partition Coefficients	Flashed tube flow – none Non-flashed tube flow – 100
Time to Reach 212 °F and Terminate Steam Release	10.32 hours
RCS Mass	Pre-accident Iodine spike – 406,715 lb _m Concurrent Iodine spike – 474,951 lb _m
SG Secondary Side Mass	minimum – 120,724 lb _m (per SG) maximum – 226,800 lb _m (per SG) Minimum used for primary-to-secondary leakage to maximize secondary nuclide concentration. Maximum used for initial secondary inventory release to maximize secondary side dose contribution.
Atmospheric Dispersion Factors Offsite Onsite	Appendix 2I Table 15.4.4-8
Control Room Ventilation System Time of Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	522.7 seconds 1.5 hours 460 cfm
Breathing Rates Offsite Control Room	RG 1.183, Section 4.1.3 RG 1.183, Section 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6

Table 15.4.4-4

SGTR Mass Release Rates

Time (hours)	Ruptured SG Break Flow Rate (lb/min)	Ruptured SG Steam Release (lb/min)	Unaffected SG Steam Release (lb/min)
0.00	2544.83	111,000	110,730
0.131	1724.28	4920	100
0.75	0.00	130	3760
1.00	26.00	130	3760
1.50	39.00	130	3760
2.00	39.00	0	3760
8.00	39.00	0	2320
12.40	0	0	0.0

Table 15.4.4-5

SGTR Iodine Equilibrium Appearance Assumptions

Input Assumption	Value
Maximum Letdown Flow	150 gpm at 120°F, 650 psia
Maximum Identified RCS Leakage	10 gpm
Maximum Unidentified RCS Leakage	1 gpm
RCS Mass	474,451 lb _m
Iodine Total Removal Coefficient (min ⁻¹)	
I-131	0.002810
I-132	0.007773
I-133	0.003305
I-134	0.015930
I-135	0.004498

Table 15.4.4-6

SGTR Concurrent Iodine Spike (335 x $\mu\text{Ci/gm}$) Activity Appearance Rate

Nuclide	Iodine Appearance (Ci/min)	8-hour Production (Ci)
I-131	170.8	82,006
I-132	94.7	45,478
I-133	207.8	99,767
I-134	88.8	42,629
I-135	127.7	61,275

Table 15.4.4-7

SGTR 60 $\mu\text{Ci/gm}$ D.E. I-131 Activities

Isotope	Activity ($\mu\text{Ci/gm}$)
I-131	50.6
I-132	10.1
I-133	52.3
I-134	4.6
I-135	23.6

Table 15.4.4-8
Control Room χ /Qs

Time (Hours)	χ/Q (sec/m³)
0	3.02E-03
0.131	6.30E-03
0.145	2.84E-03
1.5	1.62E-03
2	1.32E-03
8	5.06E-04
24	3.88E-04
96	3.30E-04
720	3.30E-04

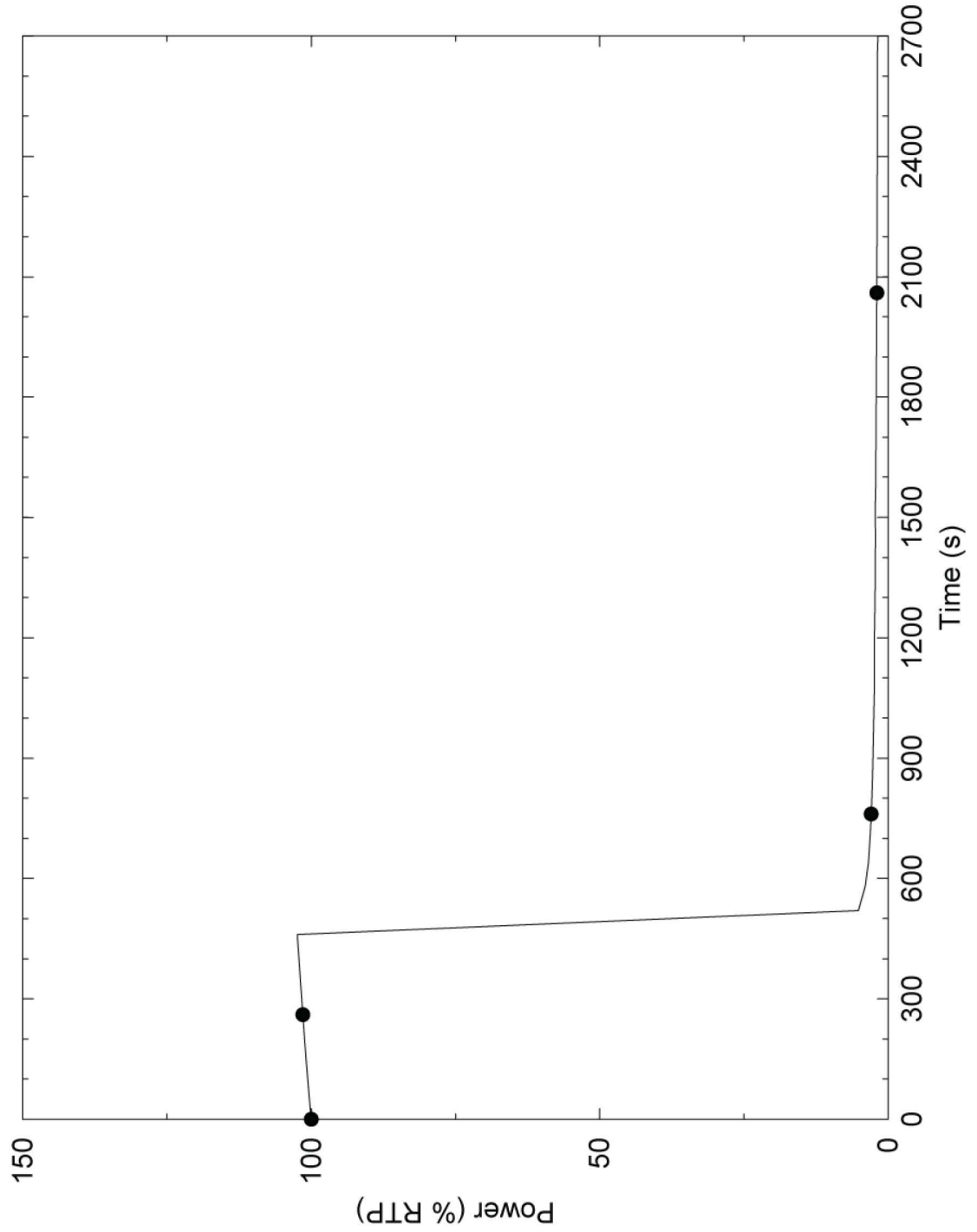
Table 15.4.4-9
SGTR Dose Consequences

Case	EAB Dose ⁽¹⁾ (REM TEDE)	LPZ Dose ⁽²⁾ (REM TEDE)	Control Room Dose ⁽²⁾ (REM TEDE)
SGTR pre-accident iodine spike	0.37	0.37	4.67
Acceptance Criteria (pre-accident iodine spike)	25 ⁽³⁾	25 ⁽³⁾	5 ⁽⁴⁾
SGTR concurrent iodine spike	0.18	0.28	2.29
Acceptance Criteria (concurrent iodine spike)	2.5 ⁽³⁾	2.5 ⁽³⁾	5 ⁽⁴⁾

- ⁽¹⁾ Worst 2-hour dose
- ⁽²⁾ Integrated 30-day dose
- ⁽³⁾ RG 1.183, Table 6
- ⁽⁴⁾ 10 CFR 50.67

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Reactor Power versus Time – Steam Generator Tube Rupture (Hot Side Break)



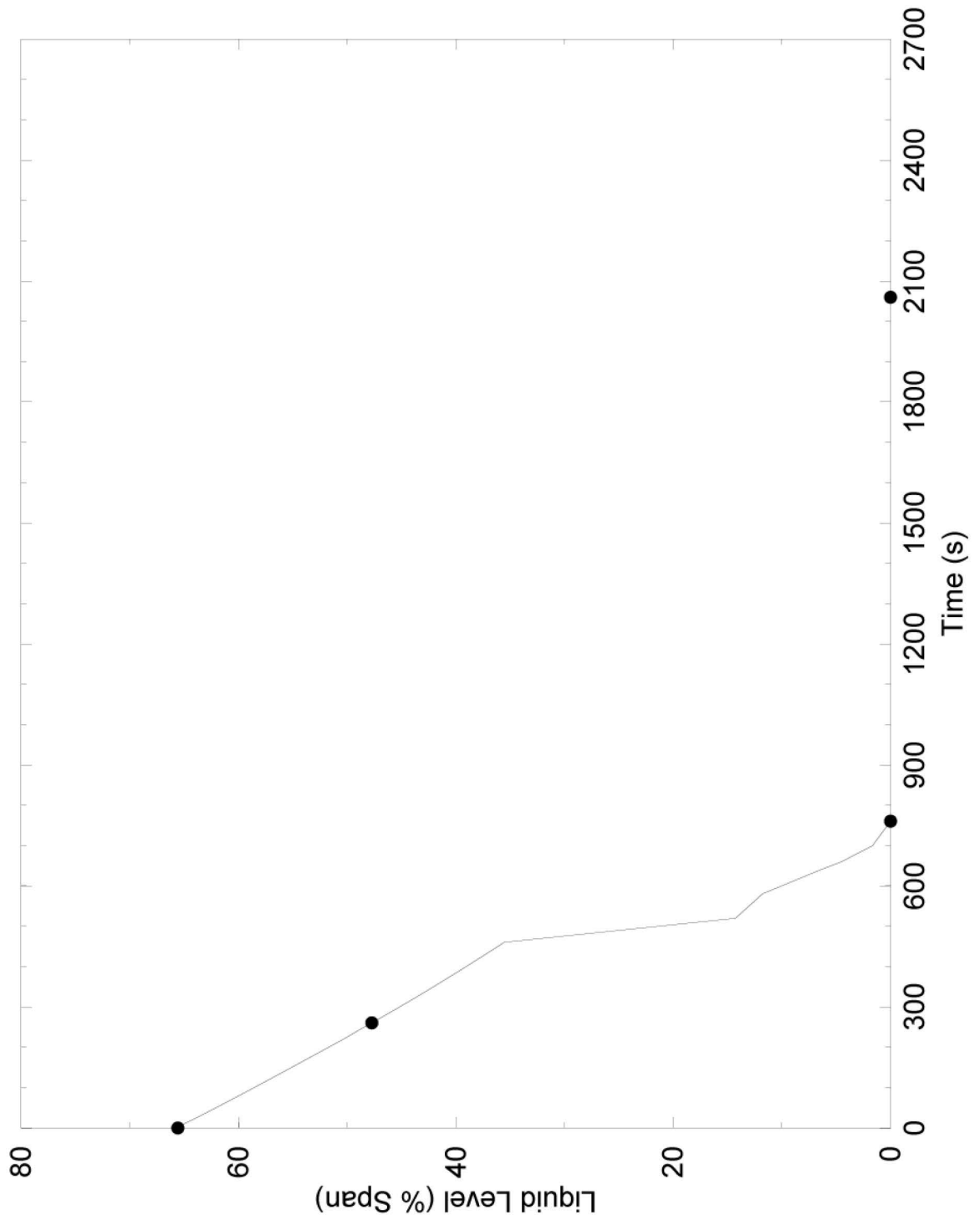
Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**REACTOR POWER VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)**

FIGURE 15.4.4-1

**Pressurizer Liquid Level versus Time
Steam Generator Tube Rupture (Hot Side Break)**

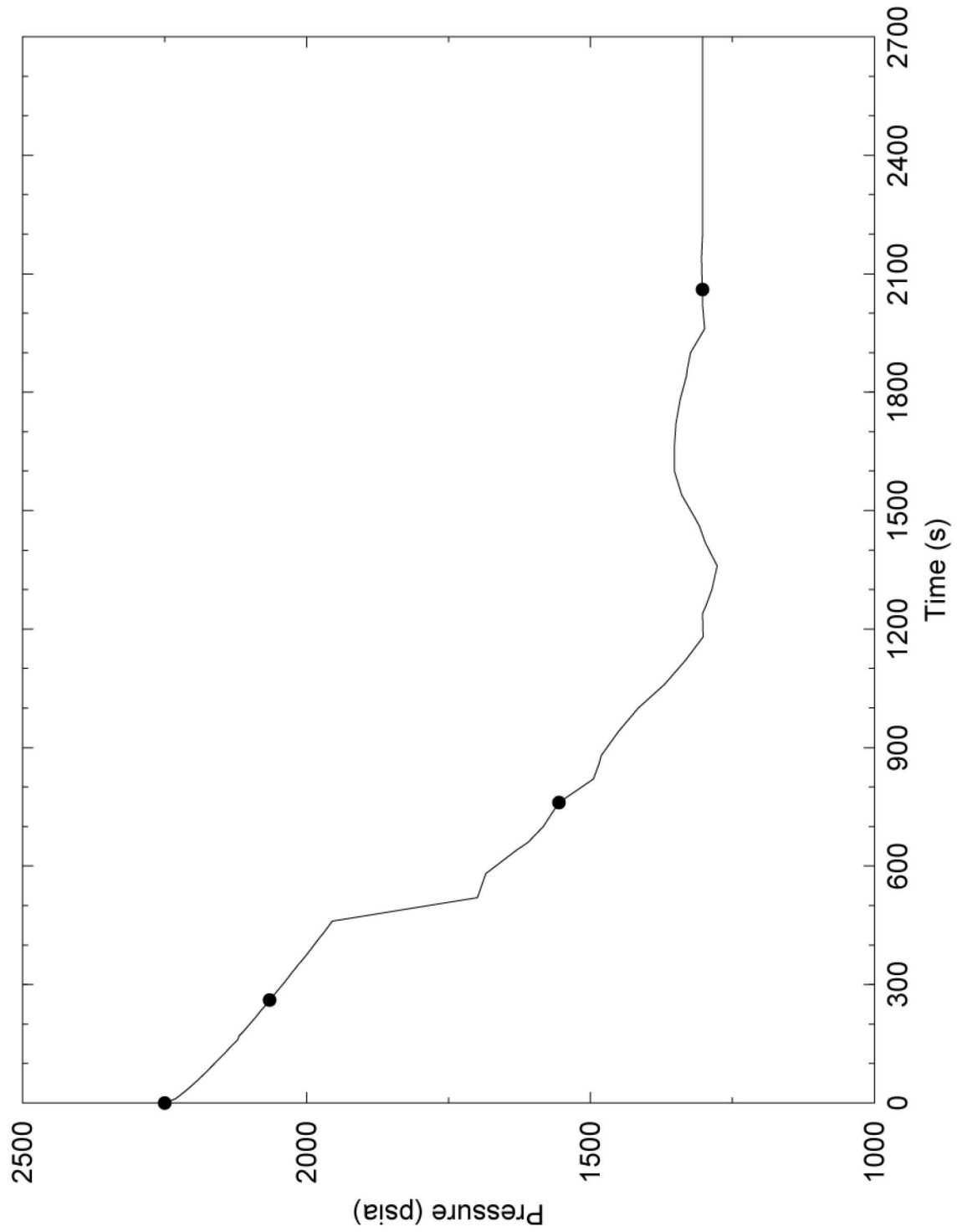


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**PRESSURIZER LIQUID LEVEL VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)
FIGURE 15.4.4-2**

**Pressurizer Pressure versus Time
Steam Generator Tube Rupture (Hot Side Break)**



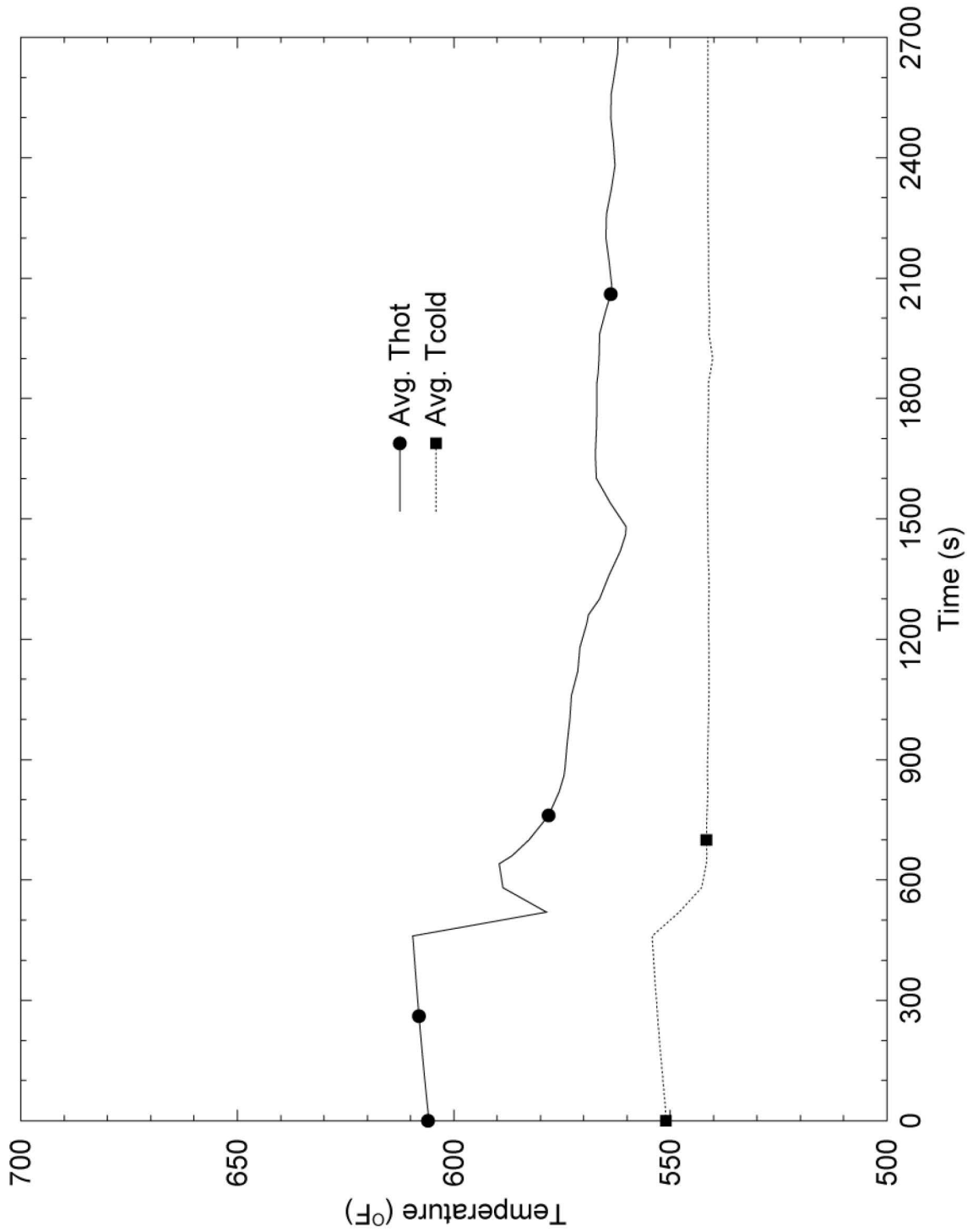
Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
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**PRESSURIZER PRESSURE VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)**

FIGURE 15.4.4-3

**RCS Loop Temperatures versus Time
Steam Generator Tube Rupture (Hot Side Break)**



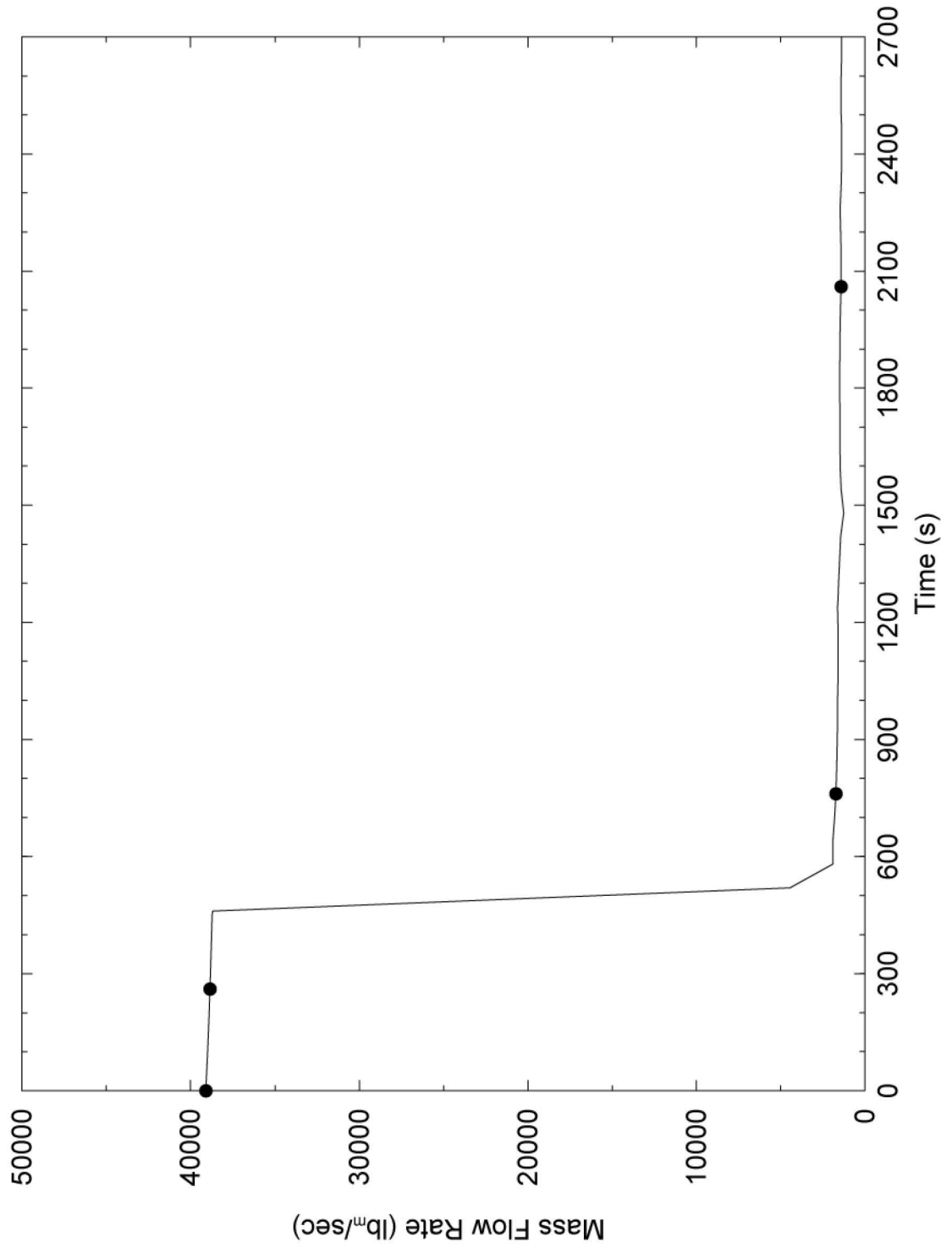
Amendment No. 26 (11/13)

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**RCS LOOP TEMPERATURES VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)**

FIGURE 15.4.4-4

**RCS Total Loop Flow Rate versus Time
Steam Generator Tube Rupture (Hot Side Break)**



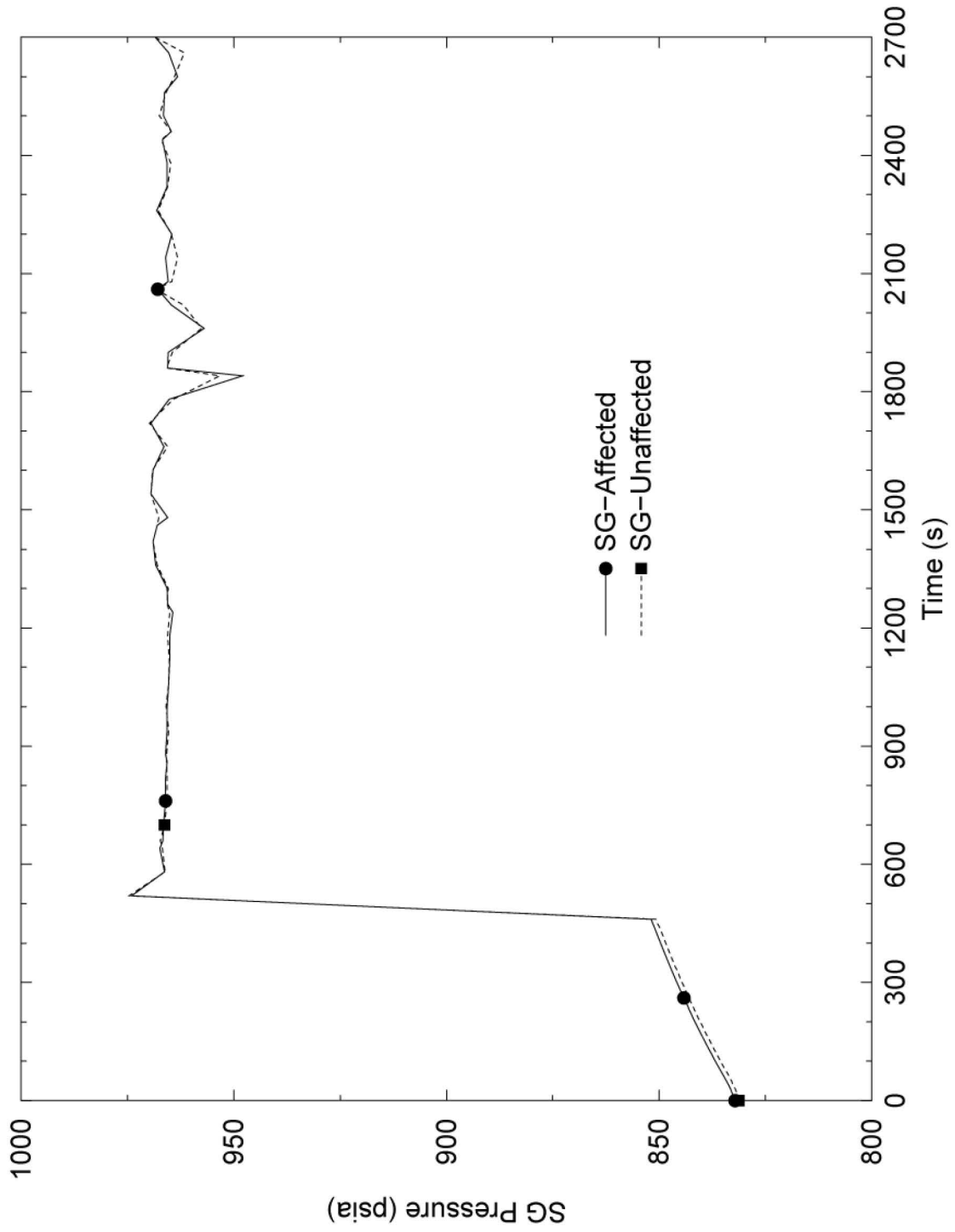
Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**RCS TOTAL LOOP FLOW RATE VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)**

FIGURE 15.4.4-5

**Steam Generator Pressures versus Time
Steam Generator Tube Rupture (Hot Side Break)**



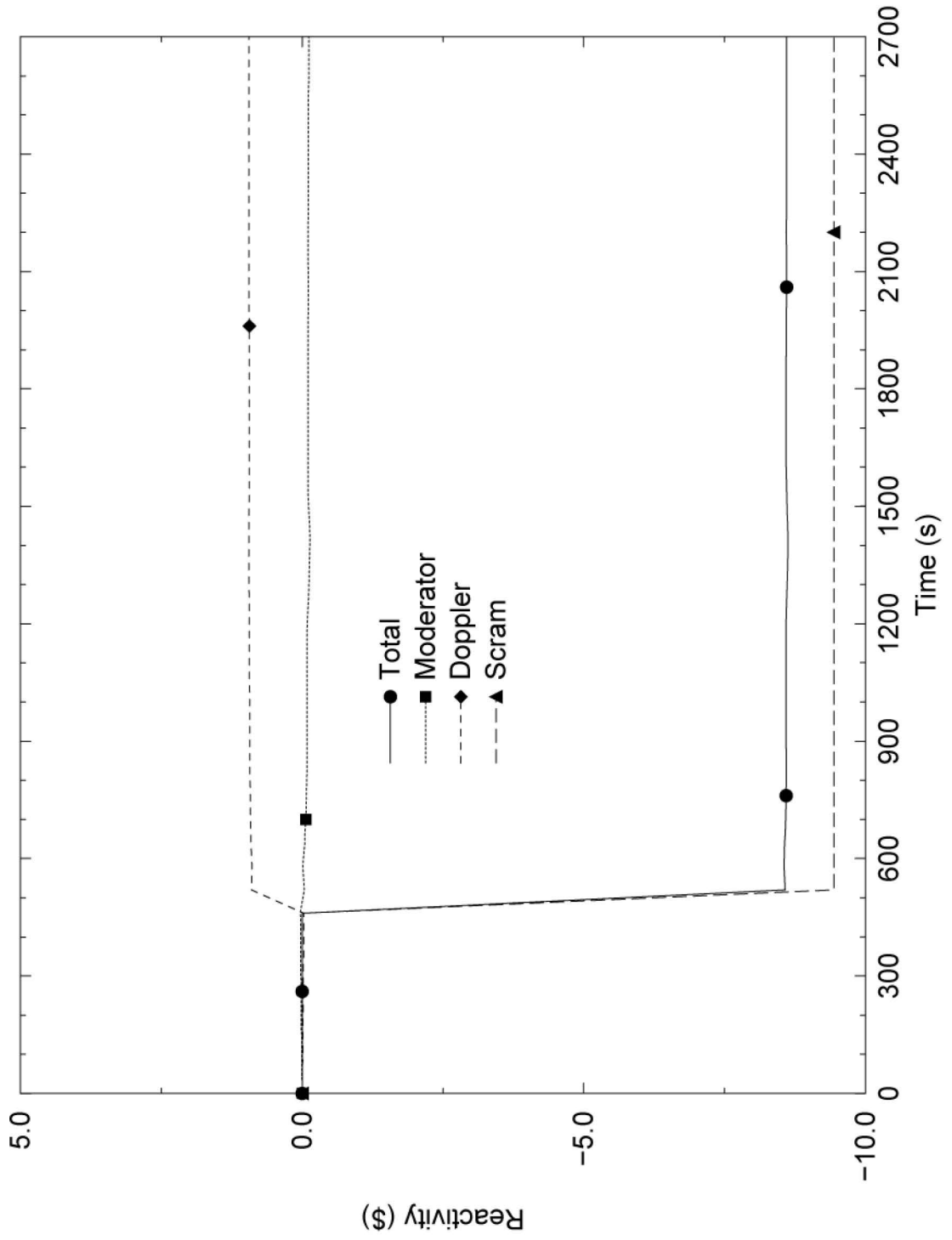
Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**STEAM GENERATOR PRESSURES
VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)**

FIGURE 15.4.4-6

**Reactivity Feedback versus Time
Steam Generator Tube Rupture (Hot Side Break)**



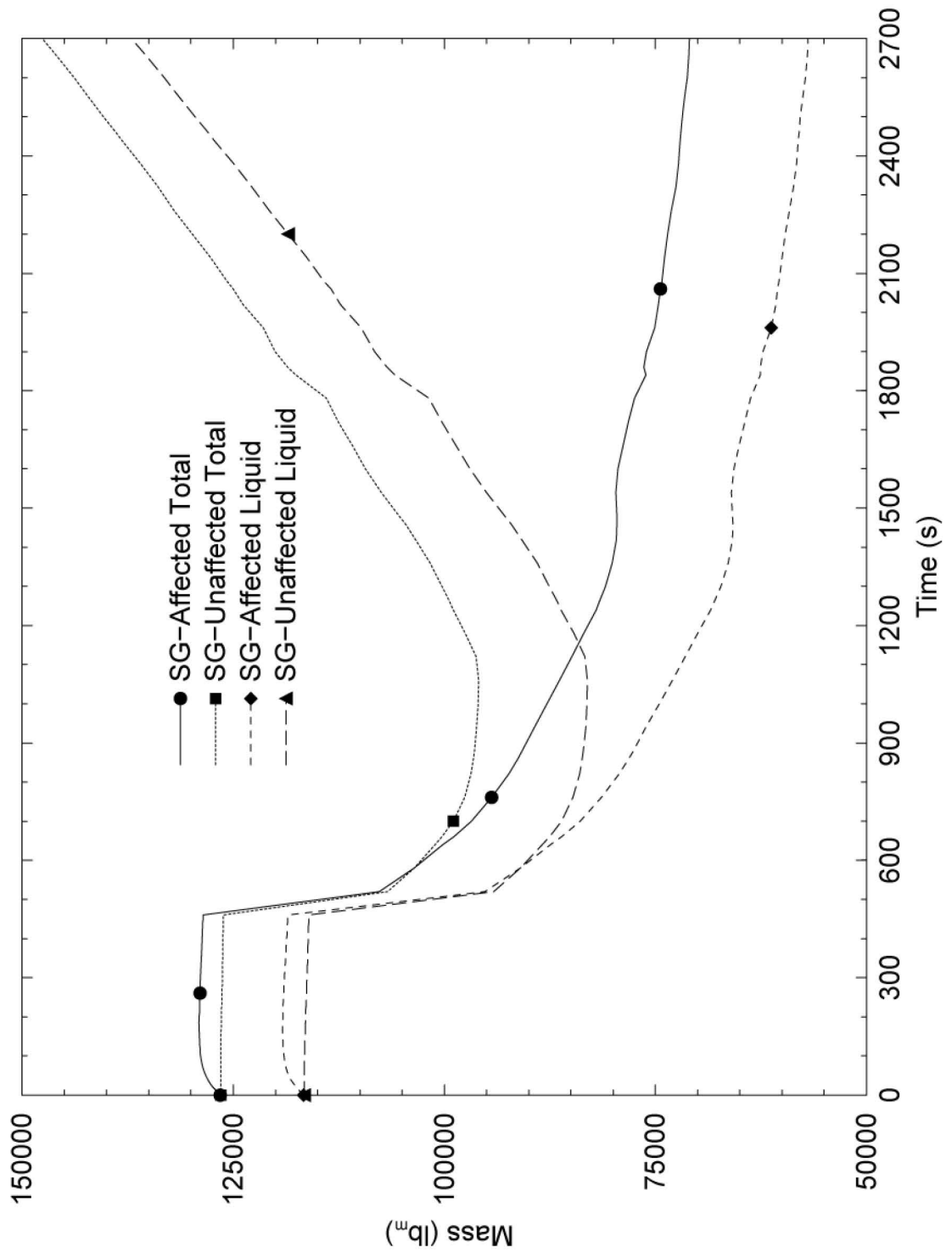
Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**REACTIVITY FEEDBACK VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)**

FIGURE 15.4.4-7

**Steam Generator Masses versus Time
Steam Generator Tube Rupture (Hot Side Break)**

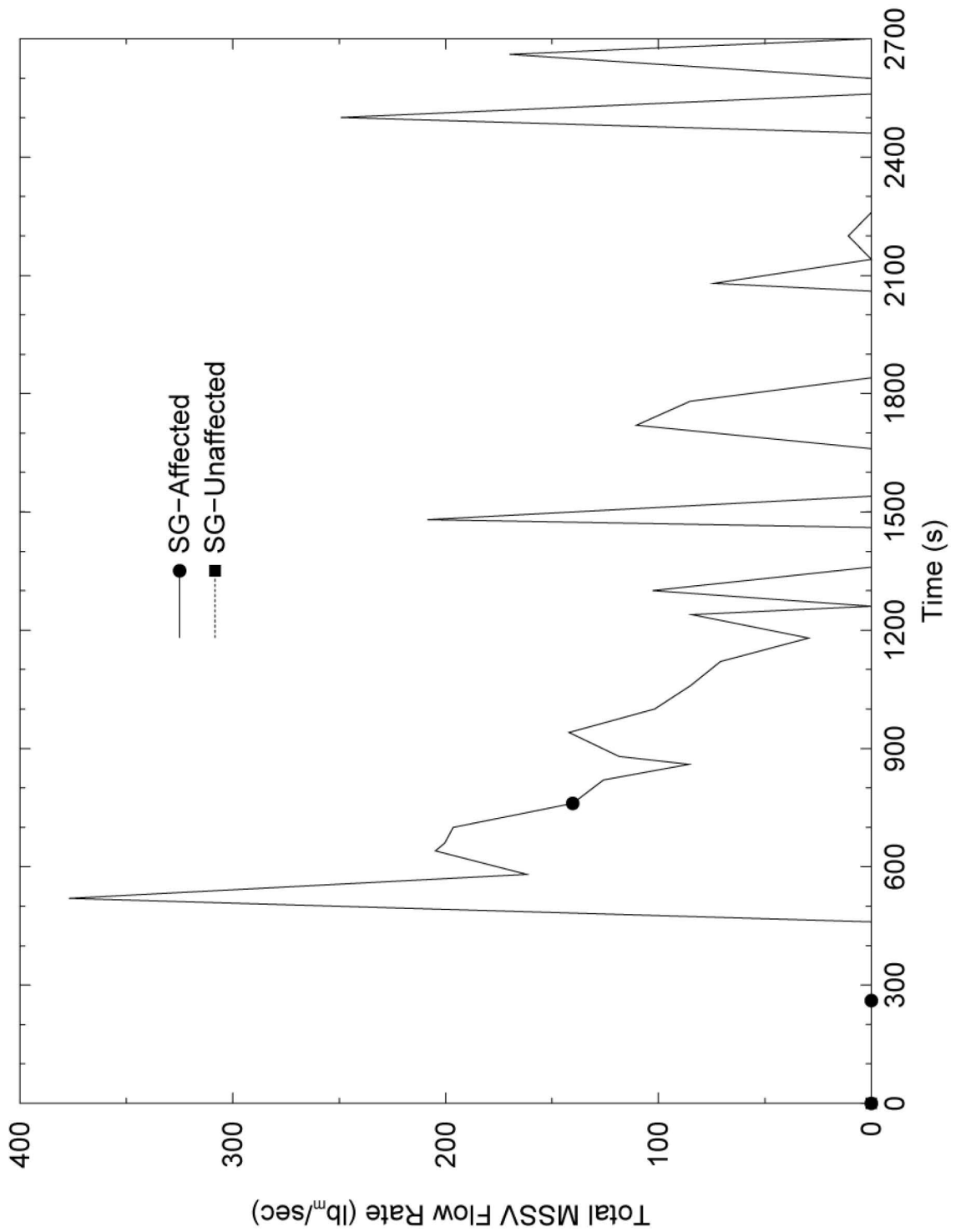


Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**STEAM GENERATOR MASSES VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)
FIGURE 15.4.4-8**

**MSSV Flow Rate versus Time
Steam Generator Tube Rupture (Hot Side Break)**



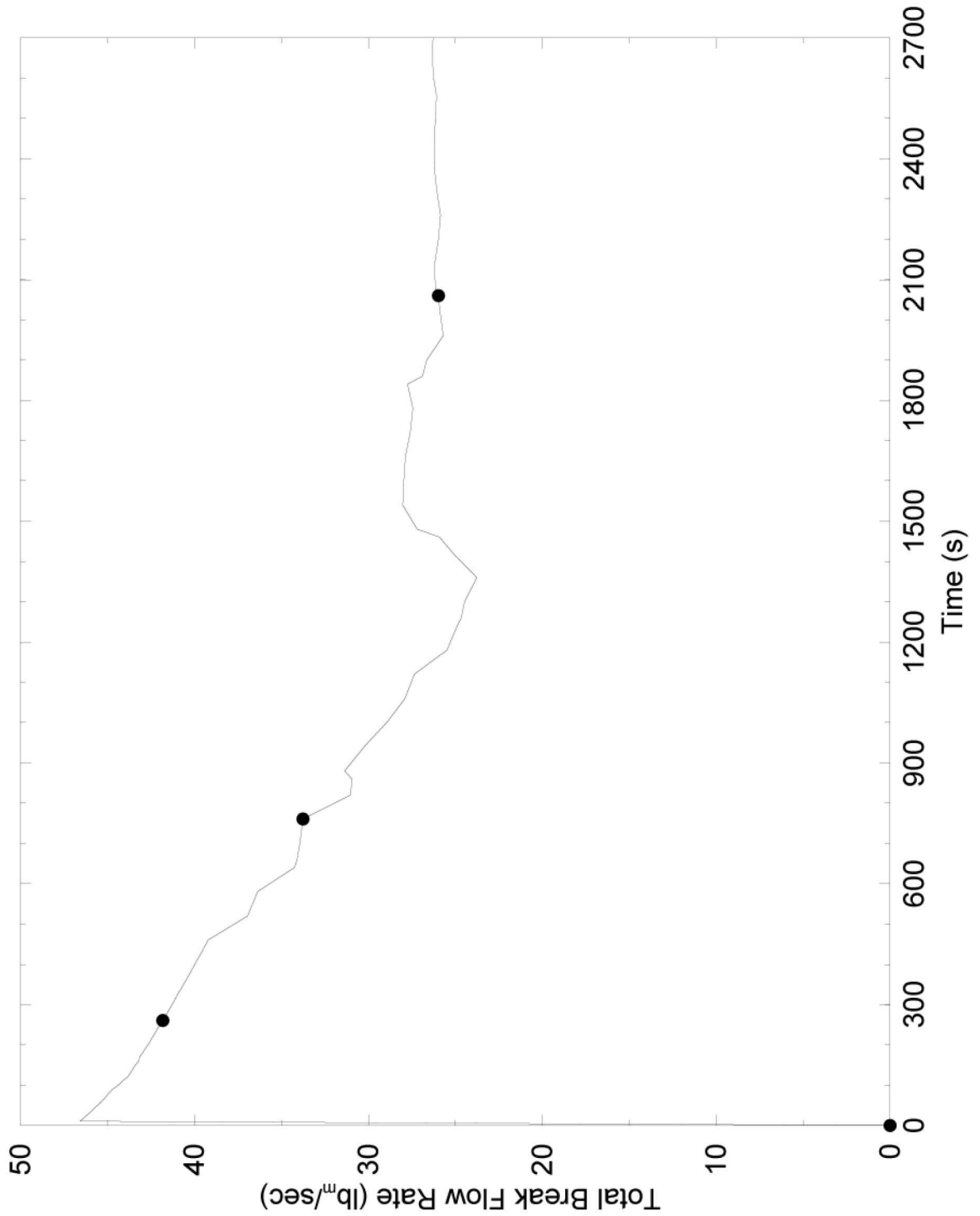
Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**MSSV FLOW RATE VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)**

FIGURE 15.4.4-9

**Total Break Flow Rate versus Time
Steam Generator Tube Rupture (Hot Side Break)**



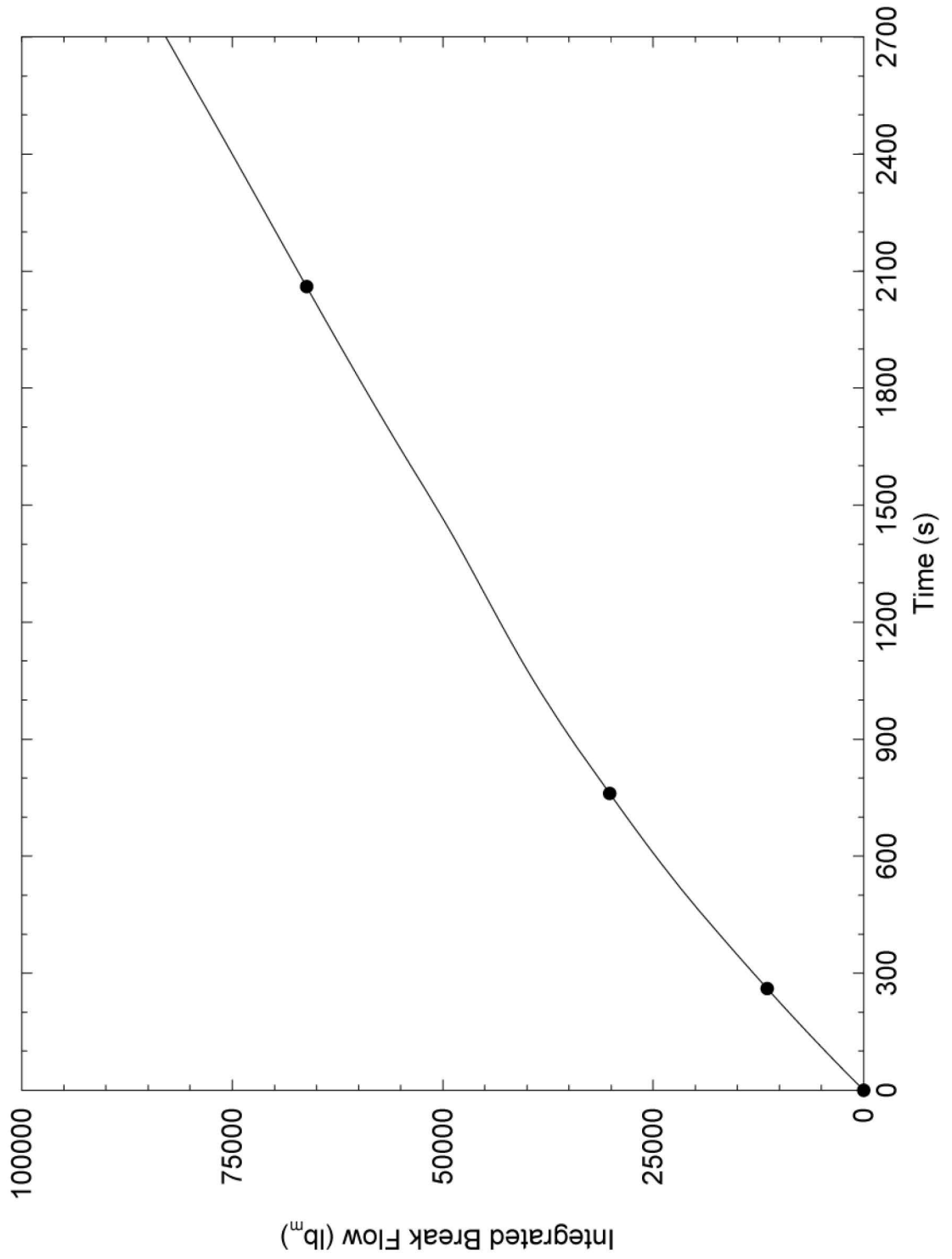
Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**TOTAL BREAK FLOW RATE VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)**

FIGURE 15.4.4-10

**Integrated Break Flow versus Time
Steam Generator Tube Rupture (Hot Side Break)**

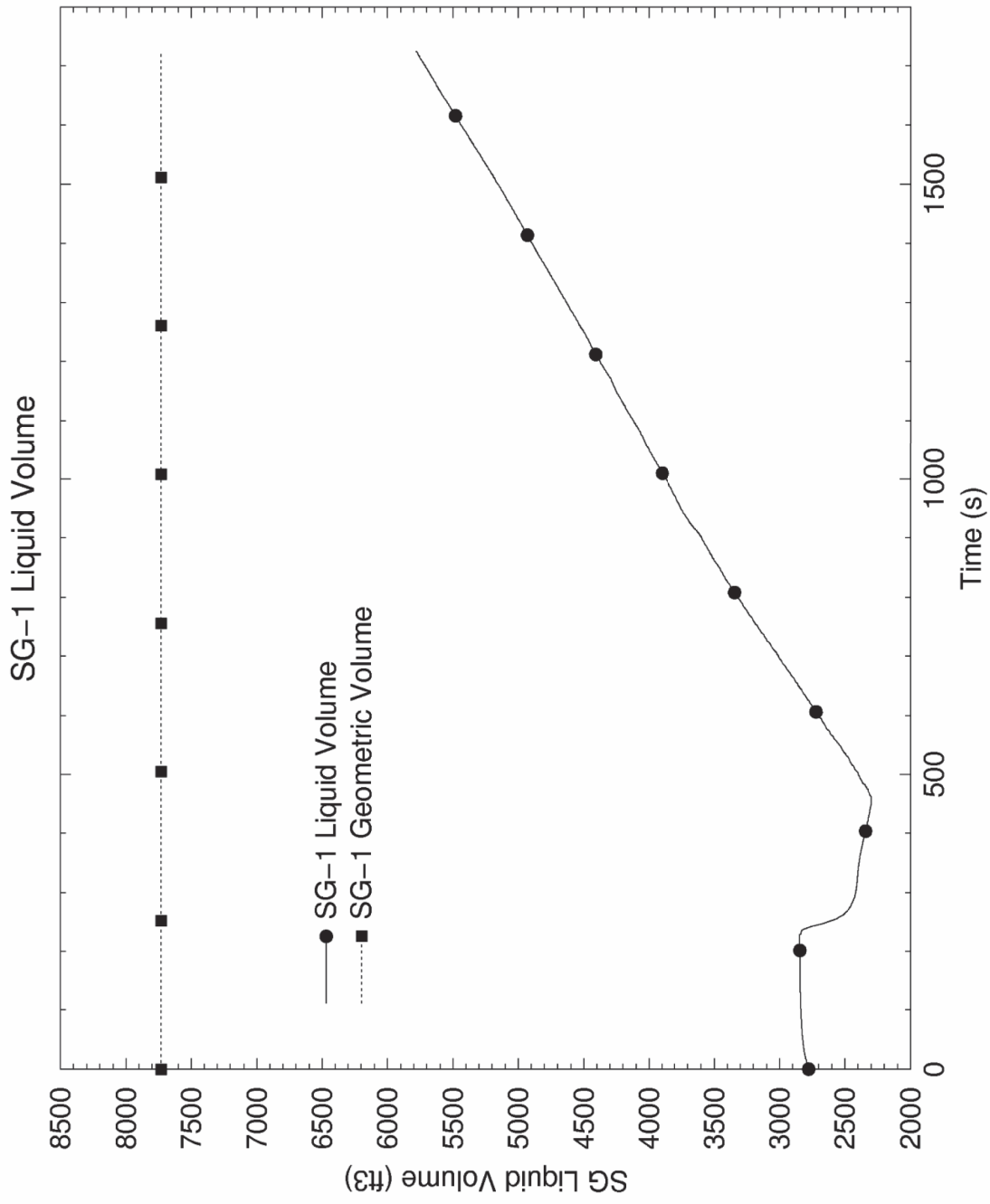


Amendment No. 26 (11/13)

**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**INTEGRATED BREAK FLOW VS TIME
STEAM GENERATOR TUBE RUPTURE
(HOT SIDE BREAK)**

FIGURE 15.4.4-11



**FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**
RUPTURED STEAM GENERATOR LIQUID
VOLUME VS TIME
STEAM GENERATOR TUBE RUPTURE (OVERFILL)
FIGURE 15.4.4-12

Amendment No. 29 (10/18)

15.4.5 CEA EJECTION EVENT

15.4.5.1 Identification of Causes

The CEA ejection event is analyzed to determine the fraction of fuel pins that exceed the criteria for clad damage.

Rapid ejection of a control element assembly (CEA) from the core would require a complete circumferential break of the control element drive mechanism (CEDM) housing or of the CEDM nozzle on the reactor vessel head. The CEDM housing and CEDM nozzle are an extension of the reactor coolant system boundary and designed and manufactured to Section III of the ASME Boiler and Pressure Vessel Code. Hence, the occurrence of such a failure is considered highly unlikely.

A typical CEA ejection transient behaves in the following manner: After ejection of a CEA from the full power or zero power (critical) initial conditions, the core power rises rapidly for a brief period. The rise is terminated by the Doppler effect. Reactor shutdown is initiated by the high power level trip, and the power transient is then completed. The core is protected against severe fuel damage by the allowable CEA patterns and by the high power trip; the maximum enthalpy in the fuel during the transient is limited to an acceptable value.

15.4.5.2 Analysis of Effects and Consequences

A CEA (or Control Rod) Ejection event is initiated by a postulated rupture of a control rod drive mechanism housing. Such a rupture allows the full system pressure to act on the drive shaft, which ejects its control rod from the core. The consequences of the mechanical failure are a rapid positive reactivity insertion and an increase in radial power peaking, which could possibly lead to localized fuel rod damage.

Doppler reactivity feedback mitigates the power excursion as the fuel begins to heat up. Although the initial increase in power occurs too rapidly for the scram rods to have any significant effect on the power during that portion of the transient, the scram reactivity does affect the fuel temperature and fuel rod cladding surface heat flux.

The ejected rod causes localized peaking such that fuel failure may occur due to DNB or FCM. The RCS pressure increases for this event which may challenge the overpressure criterion.

Detailed analyses were performed with approved methodologies using the S-RELAP5 and XCOBRA-IIIC codes (References 117 and 108). The S-RELAP5 code was used to model the key system components and calculate neutron power, fuel thermal response, surface heat transport, and fluid conditions (such as coolant flow rates, temperatures, and pressures) and produce an estimated time of MDNBR and peak system pressures. The core fluid boundary conditions and average rod surface heat flux were then input to the XCOBRA-IIIC code, which was used to calculate the MDNBR using the HTP CHF correlation (Reference 110). Table 15.4.5-3 lists assumptions used in the radiological dose calculations.

Deposited enthalpy was calculated using the Reference 111 methodology.

The rod ejection analysis was performed using both BOC and EOC initial conditions at power levels of HFP, 70% RTP, 20% RTP and HZP. Per the Technical Specifications, the core is held subcritical by more than 1% for Mode 3 (Hot Standby), Mode 4 (Hot Shutdown), and Mode 5 (Cold Shutdown). Since 1% is more than the worth of the ejected control rod, evaluation of these modes is not required. For this analysis, Hot Zero Power is therefore Mode 2 (Startup).

All four reactor coolant pumps are assumed to be in operation in both Mode 1 (Power Operation) and Mode 2 (Startup).

While this postulated event could have a failure of the reactor coolant pressure boundary, it is not clear if (or to what extent) debris pulled toward the break by fluid flow would clog or block the break.

Because of this uncertainty, conservative assumptions are typically used to bias the RCS pressure transient response. The evaluation of maximum RCS pressure for this event is based on a plugged hole in the head and takes no credit for pressure reduction from flow out of the break. For evaluation of DNB, the RCS pressure is held constant at the initial value and is assumed to neither increase nor decrease.

The input parameters and biasing for the analysis of this event is shown in Table 15.4.5-1 for the HFP and HZP cases, and in Table 15.4.5-1a for the part-power cases at 70% RTP and 20% RTP. The input parameters and biasing were consistent with the approved methodology.

- Initial Conditions - The analysis was performed from HFP, 70% RTP, 20% RTP, and HZP initial conditions to provide a bounding fuel response to the ejected CEA. Respective bounding initial fuel rod hot spot temperatures and maximum core inlet temperatures were assumed for each initial condition. Power measurement uncertainties were applied consistent with the initial power level. TS minimum RCS flow rate was modeled.
- Core Power Distributions - Conservative initial core hot spot power peaking factors corresponding to the initial power level and control rod position were used. The hot spot power peaking during the event was determined from detailed core neutronic calculations of both pre-ejection and post-ejection conditions.
- Reactivity Feedback - Reactivity feedbacks were modeled that were representative, or conservatively bounding of the BOC and EOC initial conditions. Due to the rapidity of the transient, moderator feedback has a second-order impact on the consequences. TS MTC limits were modeled for the cases initiated at BOC whereas conservatively biased "least negative" MTCs were modeled for the EOC cases. The event is initially mitigated by negative Doppler reactivity feedback. As such, the Doppler reactivity assumed in the analysis was conservatively biased to minimize the negative feedback due to increasing fuel temperatures. For the HZP initiated cases, fuel temperature dependent Doppler feedback was modeled.
- Reactor Protection System Trips and Delays - The event is primarily protected by the VHPT. The reactor protection system trip setpoints and response times were conservatively biased to delay the actuation of the trip function. The VHPT setpoints were set to values consistent with the initial power levels, including the trip uncertainty. In addition, rod insertion is delayed to account for the CEA holding coil delay time.
- Ejected CEA Worth - To maximize the core power response to the ejected CEA, a conservatively high ejected CEA worth was assumed for each case, based on St. Lucie Unit 1 specific rod patterns and power-dependent insertion limits.
- Gap Conductance - Depending on the time-in-cycle for the reactivity coefficients, gap conductance was set to either a conservative BOC value or a conservative EOC value to maximize the heat flux through the cladding and minimize the negative reactivity inserted due to Doppler feedback.

This event is classified as a Postulated Accident with the following acceptance criteria:

- Fuel failures due to DNB and FCM should be limited, so as not to impair the capability to cool the core. Additionally, the fuel failures should be within the limits of fuel failures used in the radiological analysis.
Fuel Coolability: The peak radial average fuel enthalpy should not be greater than 200 cal/gm.

- Cladding Failures: For HZP, the peak radial average fuel enthalpy should not be greater than 150 cal/gm. For intermediate power greater than 5% RTP and full power conditions, the local heat flux should not exceed thermal design limits. For pellet/cladding interaction (PCI) and pellet/cladding mechanical interaction (PCMI) failures, the change in radial average fuel enthalpy should be less than the corrosion-dependent limit depicted in Figure B-1 of SRP 4.2, Appendix B. The limit for lower burned fuel that is applicable to the EPU is for the fuel enthalpy rise to be less than 150 cal/gm. Reactivity excursions should not result in a radially averaged enthalpy greater than 200 cal/gm at any axial location in any fuel rod.
- The maximum reactor pressure during any portion of the assumed excursion should be less than the value that will cause stresses to exceed the faulted condition stress limits.
- Radiological consequences should be within the regulatory limits consistent with the design basis requirements.

15.4.5.3 Results

The results of the analysis cases are summarized in Table 15.4.5-2 for the HFP and HZP cases. The sequences of events for the HFP and HZP cases are shown in Table 15.4.5-2a. The power response for the BOC HFP case is shown in Figure 15.4.5-1. The peak hot spot centerline temperatures were compared to the respective fuel melt temperatures. MDNBR was compared to the 95/95 CHF correlation limit. Fuel failures were determined to remain within the limits of the radiological consequences analysis. The deposited enthalpies and enthalpy rise were calculated to be less than the applicable limits.

The BOC HFP case presented the most significant challenge to acceptance criteria. The transient response is shown in Figures 15.4.5-1 through 15.4.5-6. Figure 15.4.5-1 shows the reactor power as a function of time. Figure 15.4.5-2 shows the core power based on rod surface heat flux. Figures 15.4.5-3 through 15.4.5-6 show the RCS loop temperatures, the total RCS flow rate, the reactivity feedback, and the peak fuel centerline temperature, respectively.

The peak RCS pressure analysis was performed using input parameters biased to produce a conservatively high RCS pressure. The peak pressure results from the most limiting RCS pressure (BOC HFP) case was found to produce a smaller challenge to the RCS overpressure criterion than produced by the loss of external load event in Section 15.2. 7. The analysis also concluded that there is ample margin to the applicable overpressure criterion for this event, which is typically taken to be 120% of design pressure (3000 psia). The sequence of events for the overpressure analysis is provided in Table 15.4.5-2a and the plot of RCS pressure as a function of time is presented in Figure 15.4.5-7.

The calculations of record for these analyses are listed in Section 15.5, Reference (8).

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15.4.5.4 Radiological Analysis

15.4.5.4.1 Background

This event consists of an uncontrolled withdrawal of a single control element assembly (CEA). This event is the same as the Rod Ejection event referred to in RG 1.183. The CEA Ejection results in a reactivity insertion that leads to a core power level increase and subsequent reactor trip. Following the reactor trip, plant cooldown is performed using steam release from the SG ADVs. Two CEA Ejection cases are considered. The first case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere.

The second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system. The St. Lucie Unit 1 AST dose analysis methodology is presented Reference 107.

15.4.5.4.2 Compliance with RG 1.183 Regulatory Positions

The CEA Ejection dose consequence analysis followed the guidance provided in RG 1.183 Appendix H, "Assumptions for Evaluating the Radiological Consequences of a PWR Rod Ejection Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 15.4.1-1e. The inventory provided in Table 15.4.1-1e is adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The release fractions provided in RG 1.183 Table 3 are adjusted to comply with the specific RG 1.183 Appendix H release requirements. For both the containment and secondary release cases, the activity available for release from the fuel gap for fuel that experiences DNB is assumed to be 10% of the noble gas and iodine inventory in the DNB fuel. For the containment release case for fuel that experiences fuel centerline melt (FCM), 100% of the noble gas and 25% of the iodine inventory in the melted fuel is assumed to be released to the containment. For the secondary release case for fuel that experiences FCM, 100% of the noble gas and 50% of the iodine inventory in the melted fuel is assumed to be released to the primary coolant. Gap release fractions have also been increased to account for high burnup fuel rods.
2. Regulatory Position 2 - Fuel damage is assumed for this event.
3. Regulatory Position 3 - For the containment release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the containment atmosphere. For the secondary release case, 100% of the activity released from the damaged fuel is assumed to mix instantaneously and homogeneously in the primary coolant and be available for leakage to the secondary side of the SGs.
4. Regulatory Position 4 - The chemical form of radioiodine released from the damaged fuel to the containment is assumed to be 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Containment sump pH is controlled to 7.0 or higher.
5. Regulatory Position 5 - The chemical form of radioiodine released from the SGs to the environment is assumed to be 97% elemental iodine, and 3% organic iodide.
6. Regulatory Position 6.1 - For the containment leakage case, natural deposition in the containment is credited. In addition, the shield building ventilation system (SBVS) is credited. Containment spray is not credited.

7. Regulatory Position 6.2 - The containment is assumed to leak at the TS maximum allowable rate of 0.5% for the first 24 hours and 0.25% for the remainder of the event.
8. Regulatory Position 7.1 - The primary-to-secondary leak rate is apportioned between the SGs as specified by Technical Specification 5.5.6 (0.5 gpm total, 0.25 to any one SG).
9. Regulatory Position 7.2 - The density used in converting volumetric leak rates to mass leak rates is based upon RCS conditions, consistent with the plant design basis
10. Regulatory Position 7.3 - All of the noble gas released to the secondary side is assumed to be released directly to the environment without reduction or mitigation.
11. Regulatory Position 7.4 - Regulatory Position 7.4 refers to Appendix E, Regulatory Positions 5.5 and 5.6. The iodine transport model for release from the steam generators is as follows:
 - Appendix E, Regulatory Position 5.5.1 - For the secondary release case, both steam generators are used for plant cooldown. A portion of the primary-to-secondary leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncover is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.
 - Appendix E, Regulatory Position 5.5.2 - The portion of leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG, enter the steam space, and be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited.
 - Appendix E, Regulatory Position 5.5.3 - All of the SG tube leakage that does not flash is assumed to mix with the bulk water.
 - Appendix E, Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the SGs is limited by the moisture carryover from the SG. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%.
 - Appendix E, Regulatory Position 5.6 - Steam generator tube bundle uncover in the SGs is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 1. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water of the SG into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator. A conservative uncover time of 60 minutes was assumed in the analysis.

15.4.5.4.3 Other Assumptions

1. The initial RCS activity is assumed to be at the Technical Specification 3.4.16 limit of 1.0 $\mu\text{Ci/gm}$ Dose Equivalent I-131 and 518.9 $\mu\text{Ci/gm}$ DE Xe-133 gross activity. The initial SG activity is assumed to be at the Technical Specification 3.7.16 limit of 0.1 $\mu\text{Ci/gm}$ Dose Equivalent I-131.

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2. The steam mass release rates for the SGs are provided in Table 15.4.5-4.
3. The RCS fluid density used to convert the primary-to-secondary leakage from a volumetric flowrate to a mass flow rate is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. The high initial cooldown rate conservatively maximizes the fluid density. The SG tube leakage mass flow rate is provided in Table 15.4.5-5.
4. The RCS mass is assumed to remain constant throughout the event.
5. For the purposes of determining the iodine concentrations, the SG mass is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore secondary water level above the top of the tubes within a conservative time of one hour following a reactor trip.
6. Following the CEA Ejection event, 9.5% of the fuel is assumed to fail as a result of DNB and 0.5% of the fuel is assumed to experience fuel centerline melt.
7. All secondary releases are postulated to occur from the ADV with the most limiting atmospheric dispersion factors. Releases from containment through the SBVS are assumed to be released from the plant stack with a filter efficiency of 99% for particulates and 95% for both elemental and organic iodine. The activity that bypasses the SBVS is released unfiltered to the environment via a ground level release from containment.
8. The initial leakage rate from containment is 0.5% of the containment volume per day. This leak rate is reduced by 50% after 24 hours to 0.25%/day. 9.6% of the containment leakage is assumed to bypass the SBVS filters.
9. For the release inside of containment, natural deposition of the radionuclides is credited consistent with the LOCA methodology presented in Section 15.4.1.5.3. Containment sprays are not credited.
10. For the release inside of containment, containment purge is assumed coincident with the beginning of the event. As discussed in Section 6.2.5.2.2, the Hydrogen Purge system includes a demister, HEPA prefilter, two charcoal adsorber banks in series and a HEPA afterfilter. The HEPA filters are tested to meet 99.95% minimum filter efficiency. The charcoal adsorber banks are tested to meet 99% minimum filter efficiency. The analysis uses 99.5% for the HEPA filters and 98% for the charcoal filters. The Hydrogen Purge system has no automatic containment isolation valve and must be manually isolated in the event of an accident. The release fraction associated with the fuel/gap release between 30 seconds and 285 seconds when the hydrogen purge line is manually isolated is applicable.

15.4.5.4.4 Methodology

Input assumptions used in the dose consequence analysis of the CEA Ejection are provided in Table 15.4.5-3. The postulated accident consists of two cases. One case assumes that 100% of the activity released from the damaged fuel is instantaneously and homogeneously mixed throughout the containment atmosphere, and the second case assumes that 100% of the activity released from the damaged fuel is completely dissolved in the primary coolant and is available for release to the secondary system.

For the containment release case, 100% of the activity is released instantaneously to the containment. The releases from the containment correspond to the same leakage points discussed for the LOCA in Section 15.4.1.5.3. Natural deposition of the released activity inside of containment is credited. In addition, the shield building ventilation system (SBVS) is credited. Removal of activity via containment spray is not credited.

For the secondary release case, primary coolant activity is released into the SGs by leakage across the SG tubes. The activity on the secondary side is then released via steaming from the ADVs until the RCS is cooled to 212°F. All noble gases associated with this leakage are assumed to be released directly to the environment. The primary-to-secondary leakage is assumed to continue until the faulted steam generator is completely isolated at 12 hours. In addition, the analysis assumes that the initial iodine activity of both SGs is immediately released to the environment. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 $\mu\text{Ci/gm}$ DE I-131 permitted by TS. These release assumptions are consistent with the requirements of RG 1.183.

The CEA Ejection is evaluated with the assumption that 0.5% of the fuel experiences FCM and 9.5% of the fuel experiences DNB. The activity released from the damaged fuel corresponds to the requirements set out in Regulatory Position 1 of Appendix H to RG 1.183. A radial peaking factor of 1.70 is applied in the development of the source terms.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 460 cfm of unfiltered inleakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 50-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 460 cfm of unfiltered inleakage, and 1760 cfm of filtered recirculation flow.
- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of up to 504 cfm of filtered makeup flow, 460 cfm of unfiltered inleakage, and 1256 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% elemental iodine, and 95% organic iodine.

15.4.5.4.5 Radiological Consequences

The Control Room atmospheric dispersion factors (χ/Q_s) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake.

For the CEA secondary side release case, releases from the SGs are assumed to occur from the ADV that produces the most limiting χ/Q_s . When the Control Room Ventilation System is in normal mode, the most limiting χ/Q corresponds to the worst air intake to the control room. When the ventilation system is isolated, the limiting χ/Q corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. Development of control room atmospheric dispersion factors is discussed in Appendix 2J. The χ/Q_s for the secondary releases are summarized in Table 15.4.5-6. For the CEA inside of containment release case, the χ/Q_s for containment leakage are assumed to be identical to those for the LOCA discussed in Section 15.4.1.5.4.

For the EAB dose analysis, the χ/Q factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period. The LPZ dose is determined using the χ/Q factors for the appropriate time intervals. These χ/Q factors are provided in Appendix 2I.

The radiological consequences of the CEA Ejection are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. As shown in Table 15.4.5-7, the results of both cases for EAB dose, LPZ dose, and Control Room dose are all within the appropriate regulatory acceptance criteria.

The calculation of record for this analysis is in Section 15.5, Reference (8).

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TABLE 15.4.5-1

CEA EJECTION: INPUT PARAMETER BIASING FOR HFP AND HZP CASES

PARAMETER	HZP		HFP	
	BOC	EOC	BOC	EOC
Core Power	3W	3W	3,020 MWt + 0.3%	3,020 MWt + 0.3%
Core Inlet Temperature	532°F	532°F	551°F	551°F
RCS Flow Rate	375,000 gpm	375,000 gpm	375,000 gpm	375,000 gpm
Pressurizer Pressure	2,250 psia	2,250 psia	2,250 psia	2,250 psia
Pressurizer Level	33.09%	33.09%	65.6% (SAFDL) 69% (overpressure)	65.6%
Scram Reactivity	3600 pcm	3600 pcm	5000 pcm	5000 pcm
Moderator Temperature Coefficient	+7 pcm/°F	-7.75 pcm/°F	+2 pcm/°F	-7.75 pcm/°F
Doppler Reactivity Coefficient	BOC fuel temperature dependent feedback	EOC fuel temperature dependent feedback	-0.80 pcm/°F	-1.00 pcm/°F
Gap Conductance	Conservative BOC value	Conservative EOC value	Conservative BOC value	Conservative EOC value
VHP RPS Trip Setpoint	25% RTP	25% RTP	112% RTP	112% RTP
VHP RPS Trip Delay	1.1 seconds	1.1 seconds	0.4 seconds	0.4 seconds
CEA Holding Coil Delay	0.5 seconds	0.5 seconds	0.5 seconds	0.5 seconds
CEA Drop Time (excluding holding coil delay time)	2.9 seconds	2.9 seconds	2.9 seconds	2.9 seconds
Ejected CEA worth	650 pcm	530 pcm	400 pcm	150 pcm
Post-ejection F_Q	5.0	12.0	2.5	2.5
SG Tube Plugging	10%	10%	10%	10%

TABLE 15.4.5-1a

CEA EJECTION: INPUT PARAMETER BIASING FOR PART-POWER CASES

PARAMETER	20% RTP	20% RTP	70% RTP	70% RTP
	BOC	EOC	BOC	EOC
Ejected rod worth, pcm	525	350	375	150
Doppler Reactivity Coefficient, pcm/°F	-1.21	-1.39	-1.12	-1.31
Post ejection F_Q	4.49	7.60	3.58	3.33
Delayed neutron fraction	0.00638	0.00527	0.00638	0.00527

TABLE 15.4.5-2

CEA EJECTION: EVENT RESULTS FOR HFP AND HZP CASES

EVENT DESCRIPTION	RESULT PARAMETER	ANALYSIS LIMIT	ANALYSIS RESULT*
CEA EJECTION			
• BOC, HZP	MDNBR (% fuel failure)	≥ 1.164	2.442 (0%)
	Max. centerline temperature, °F (% fuel failure)	$\leq 4,623^*$	4,038 (0%)
	Peak radial average enthalpy, cal/gm	≤ 150	26.4
	Peak radial average fuel enthalpy rise, cal/gm	≤ 150 for lower burned fuel	100
• EOC, HZP	MDNBR (% fuel failure)	≥ 1.164	2.917 (0%)
	Max. centerline temperature, °F (% fuel failure)	$\leq 4,623^*$	3,212 (0%)
	Peak radial average enthalpy, cal/gm	≤ 150	21.6
	Peak radial average fuel enthalpy rise, cal/gm	≤ 150 for lower burned fuel	100
• BOC, HFP	MDNBR (% fuel failure)	≥ 1.164	1.234 (0%)
	Max. centerline temperature, °F (% fuel failure)	$\leq 4,623^*$	4,607 (0%)
	Peak radial average enthalpy, cal/gm	≤ 200	190.9
	Peak radial average fuel enthalpy rise, cal/gm	≤ 150 for lower burned fuel	100
	Max. RCS pressure, psia	$\leq 2,750$	2,696
• EOC, HFP	MDNBR (% fuel failure)	≥ 1.164	1.984 (0%)
	Max. centerline temperature, °F (% fuel failure)	$\leq 4,623^*$	4,385 (0%)
	Peak radial average enthalpy, cal/gm	≤ 200	185.8
	Peak radial average fuel enthalpy rise, cal/gm	≤ 150 for lower burned fuel	100

* These values will change from cycle to cycle. The event is re-analyzed each cycle to verify that the fuel failure limits are met if the MDNBR is below the 95/95 CHF correlation limit.

TABLE 15.4.5-2a

CEA EJECTION: SEQUENCE OF EVENTS FOR HFP AND HZP CASES

CASE	EVENT	TIME (SEC.)	
		SAFDL	Over-pressure
BOC HZP	Beginning of reactivity insertion	0.0	
	Ejected CEA fully withdrawn	0.10	
	VHPT setpoint reached (NI signal)	1.18	
	Maximum nuclear power	1.42	
	Reactor scram VHPT (including trip response delay)	2.28	
	CEA insertion begins	2.78	
	Maximum core heat flux through cladding	4.3	
	MDNBR	4.3	
	Maximum fuel centerline temperature	5.4	
EOC HZP	Beginning of reactivity insertion	0.0	
	Ejected CEA fully withdrawn	0.10	
	VHPT setpoint reached (NI signal)	1.44	
	Maximum nuclear power	1.62	
	Reactor scram VHPT (including trip response delay)	2.54	
	CEA insertion begins	3.04	
	MDNBR	4.0	
	Maximum core heat flux through cladding	4.1	
	Maximum fuel centerline temperature	5.4	
BOC HFP	Beginning of reactivity insertion	0.0	0.0
	VHPT setpoint reached (NI signal)	0.02	0.02
	Ejected CEA fully withdrawn	0.10	0.10
	Maximum nuclear power	0.14	0.14
	Reactor scram VHPT (including trip response delay)	0.42	0.42
	CEA insertion begins	0.92	0.92
	Maximum core heat flux through cladding	2.0	2.0
	MDNBR	2.0	N/A
	Pressurizer safety valves open	N/A	3.3
	Maximum fuel centerline temperature	3.3	N/A
	Maximum RCS pressure	N/A	3.5
	Pressurizer safety valves close	N/A	4.6
EOC HFP	Beginning of reactivity insertion	0.0	
	VHPT setpoint reached (NI signal)	0.05	
	Ejected CEA fully withdrawn	0.10	
	Maximum nuclear power	0.13	
	Reactor scram VHPT (including trip response delay)	0.45	
	CEA insertion begins	0.95	
	Maximum core heat flux through cladding	1.3	
	MDNBR	1.3	
Maximum fuel centerline temperature	3.0		

TABLE 15.4.5-2b

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Table 15.4.5-3
Control Element Assembly (CEA) Ejection – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3030 MW _{th} (3020 + 0.3%)
Core Average Fuel Burnup	49,000 MWD/MTU
Fuel Enrichment	1.5 - 5.0 w/o
Maximum Radial Peaking Factor	1.65
Percent of Fuel Rods in DNB	9.5%
Percent of Fuel Rods with Centerline Melt	0.5%
Core Fission Product Inventory	Table 15.4.1-1e
Initial RCS Equilibrium Activity	1.0 μCi/gm DE I-131 and 518.9 μCi/gm DE Xe-133 gross activity (Table 15.4.1-9)
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131 (Table 15.4.6-8)
Release Fraction from DNB Fuel Failures	Section 1 of Appendix H to RG 1.183
Release Fraction from Centerline Melt Fuel Failures	Section 1 of Appendix H to RG 1.183
Steam Generator Tube Leakage	0.5 gpm (Table 15.4.5-5)
Time to Terminate SG Tube Leakage	12.4 hours
Secondary Side Mass Releases to Environment	Table 15.4.5-4
SG Tube Uncovery Following Reactor Trip Time to tube recovery Flashing Fraction	1 hour 5 %
Steam Generator Secondary Side Partition Coefficient	Flashed tube flow – none Non-flashed tube flow – 100
Time to Reach 212 °F and Terminate Steam Release	12.4 hours
RCS Mass	minimum – 406,715 lb _m Minimum mass used for fuel failure dose contribution to maximum SG tube leakage activity
SG Secondary Side Mass	minimum – 120,724 lb _m (per SG) maximum – 226,800 lb _m (per SG) Minimum used for primary-to-secondary leakage to maximize secondary nuclide concentration. Maximum used for initial secondary inventory release to maximize secondary side dose contribution.
Chemical Form of Iodine Released to Containment	Particulate – 95% Elemental – 4.85% Organic – 0.15%
Chemical Form of Iodine Released from SGs	Particulate – 0% Elemental – 97% Organic – 3%

Table 15.4.5-3 (Cont'd)

Control Element Assembly (CEA) Ejection – Inputs and Assumptions

Atmospheric Dispersion Factors Offsite Onsite	Appendix 2I Table 15.4.5-6 and Appendix 2J
Control Room Ventilation System Time of Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	50 seconds 1.5 hours 460 cfm
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6
Containment Volume Containment Leakage Rate 0 to 24 hours after 24 hours	2.506E+06 ft ³ 0.5% (by volume)/day 0.25% (by volume)/day
Secondary Containment Filter Efficiency	Particulate – 99% Elemental – 95% Organic – 95%
Secondary Containment Drawdown Time	310 seconds
Secondary Containment Bypass Fraction	9.6%
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr ⁻¹ Elemental Iodine – 2.89 hr ⁻¹ Organic Iodine – None

TABLE 15.4.5-4

CEA STEAM RELEASE RATE

Time (hr)	SG Steam Release Rate (lb _m /min)
0	5486.15
0.50	2820.86
2.00	2846.36
12.40	0.00

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CEA STEAM GENERATOR TUBE LEAKAGE

Time (hr)	SG Tube Leakage (lb _m /min)
0	3.103
0.50	3.361
0.75	3.428
1.00	3.536
1.39	3.565
2.00	3.657
4.00	3.756
8.00	3.945
10.50	4.012
12.40	0.000

TABLE 15.4.5-6

CONTROL ROOM χ/Q (FOR RELEASE FROM SGs)

Time (hours)	χ/Q (sec/m ³)
0	6.30E-03
0.013889	2.84E-03
1.5	1.62E-03
2	1.32E-03
8	5.06E-04
24	3.88E-04
96	3.30E-04
720	3.30E-04

TABLE 15.4.5-7

CEA EJECTION DOSE CONSEQUENCES

Case	EAB Dose ⁽¹⁾ (REM TEDE)	LPZ Dose ⁽²⁾ (REM TEDE)	Control Room Dose ⁽²⁾ (REM TEDE)
CEA Ejection – Secondary Release	0.29	0.71	3.26
CEA Ejection – Containment Release	0.28	0.55	3.30
Acceptance Criteria	6.3	6.3	5

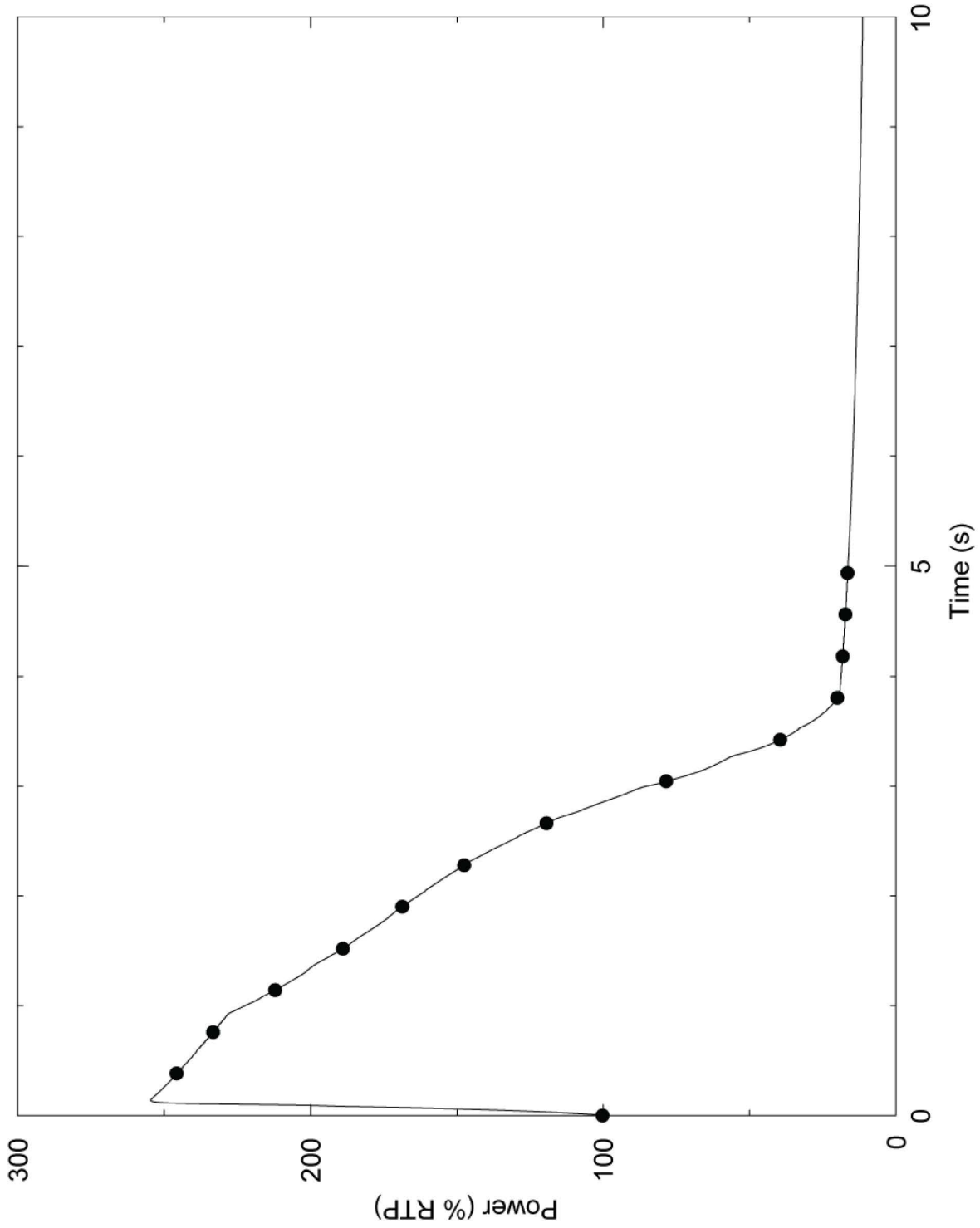
⁽¹⁾ Worst 2-hour dose

⁽²⁾ Integrated 30-day dose

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Reactor Power versus Time – CEA Ejection (BOC HFP)

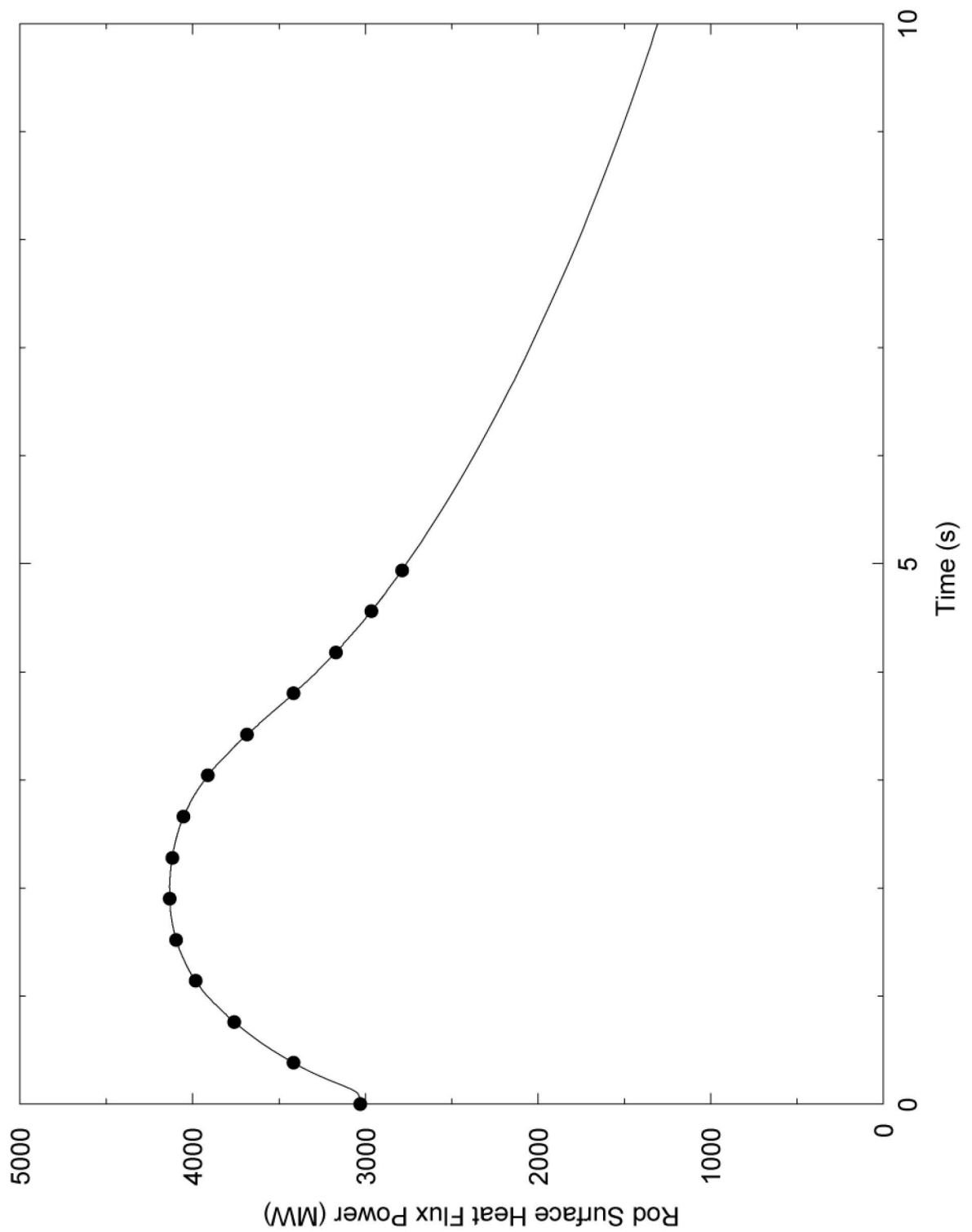


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**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1
REACTOR POWER VS TIME
CEA EJECTION (BOC HFP)**

FIGURE 15.4.5-1

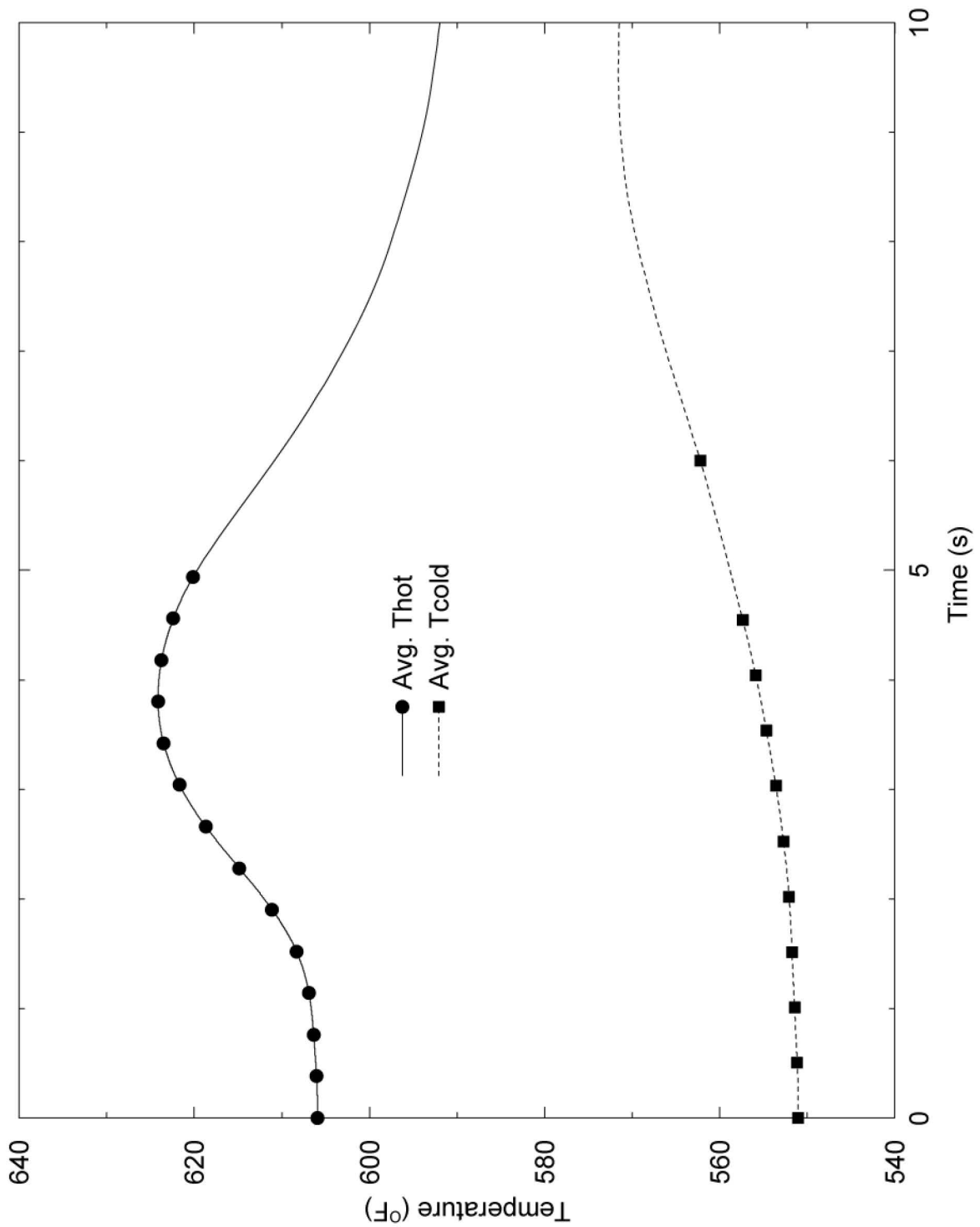
Total Core Heat Flux Power versus Time – CEA Ejection (BOC HFP)



Amendment No. 26 (11/13)

<p>FLORIDA POWER AND LIGHT COMPANY ST.LUCIE PLANT UNIT 1 TOTAL CORE HEAT FLUX POWER VS TIME CEA EJECTION (BOC HFP)</p>
<p>FIGURE 15.4.5-2</p>

RCS Loop Temperature versus Time – CEA Ejection (BOC HFP)



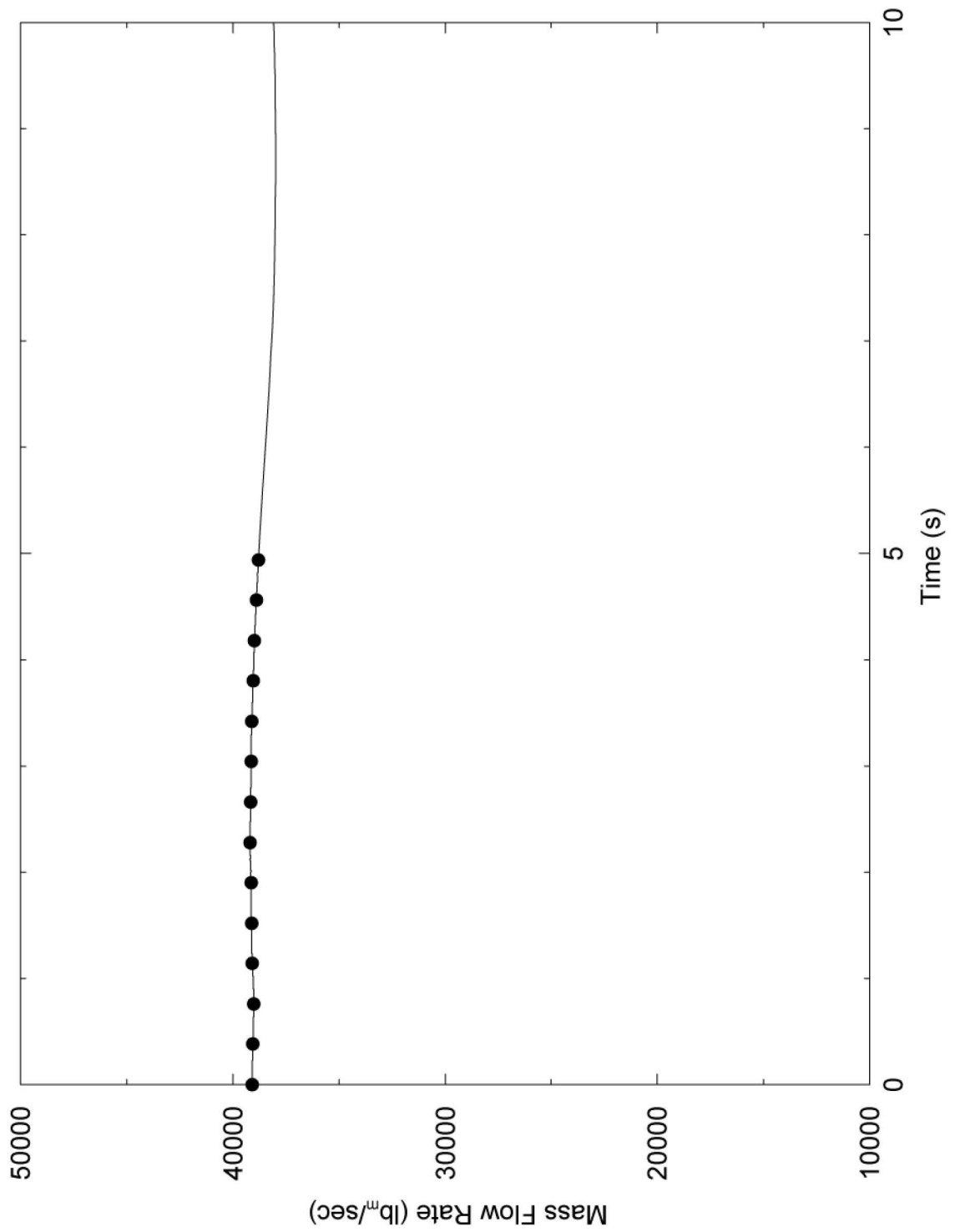
Amendment No. 26 (11/13)

FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1

RCS LOOP TEMPERATURE VS TIME
CEA EJECTION (BOC HFP)

FIGURE 15.4.5-3

RCS Total Loop Flow Rate versus Time – CEA Ejection (BOC HFP)



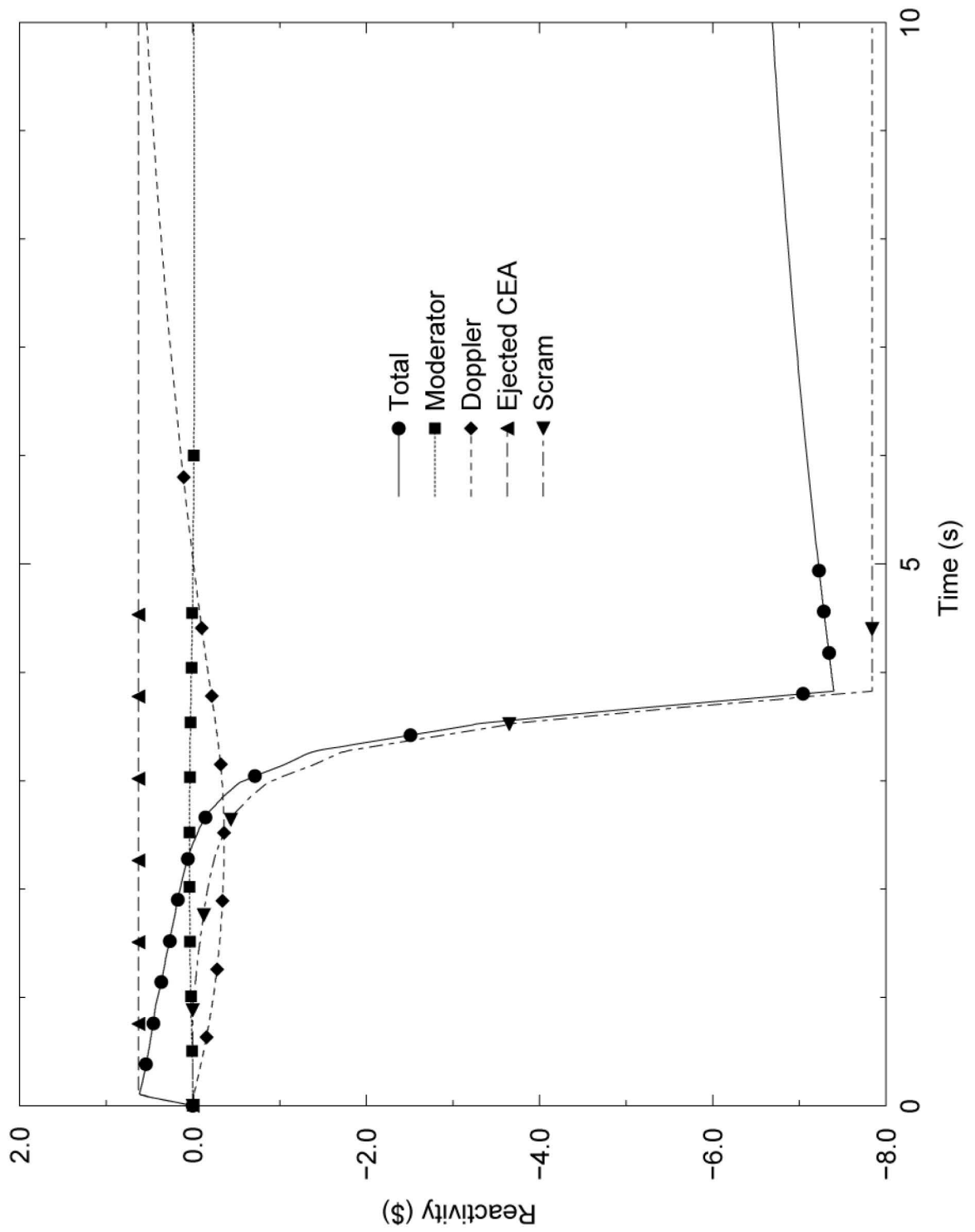
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
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**RCS TOTAL LOOP FLOW RATE VS TIME
CEA EJECTION (BOC HFP)**

FIGURE 15.4.5-4

Reactivity Feedback versus Time – CEA Ejection (BOC HFP)



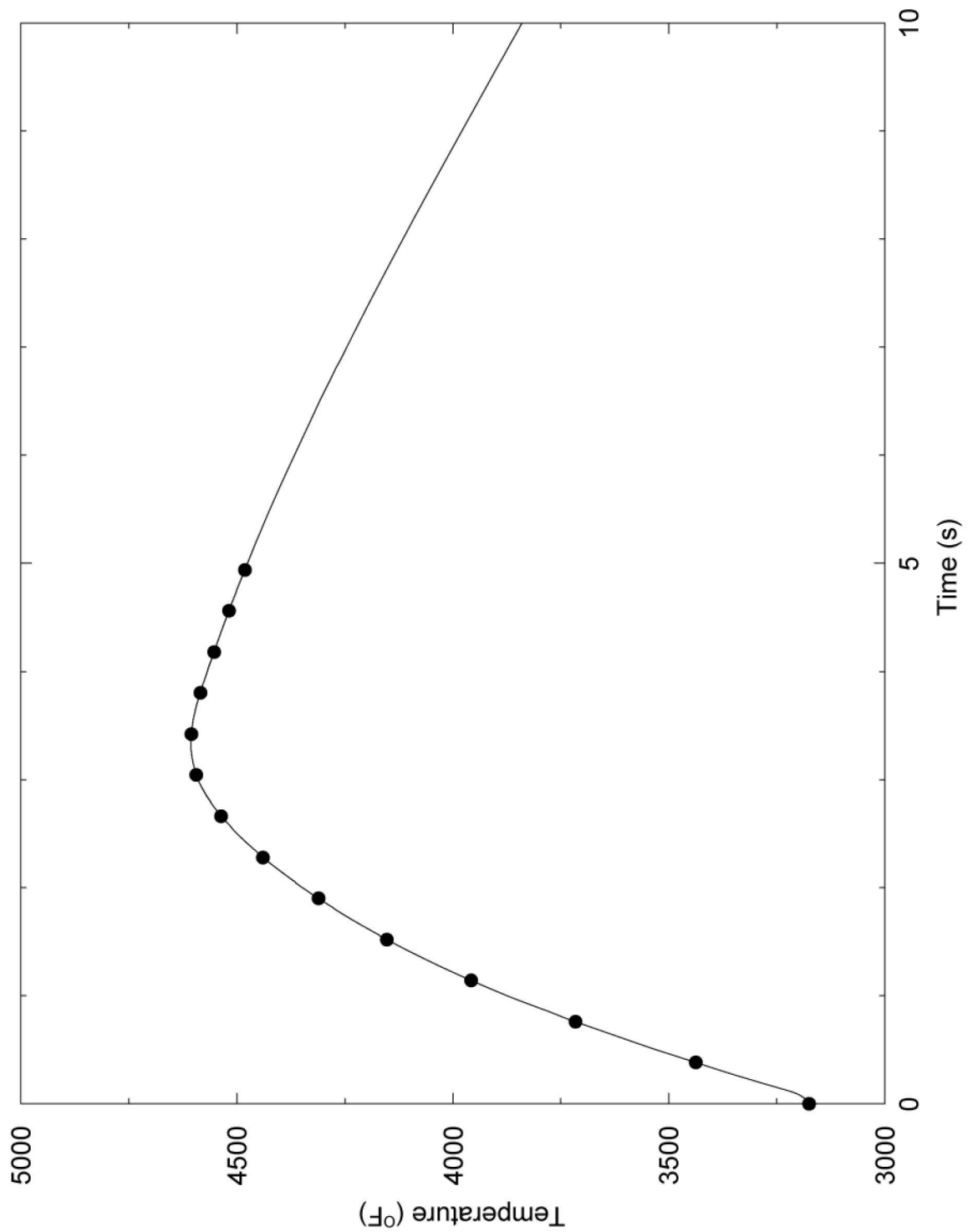
Amendment No. 26 (11/13)

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ST.LUCIE PLANT UNIT 1

REACTIVITY FEEDBACK VS TIME
CEA EJECTION (BOC HFP)

FIGURE 15.4.5-5

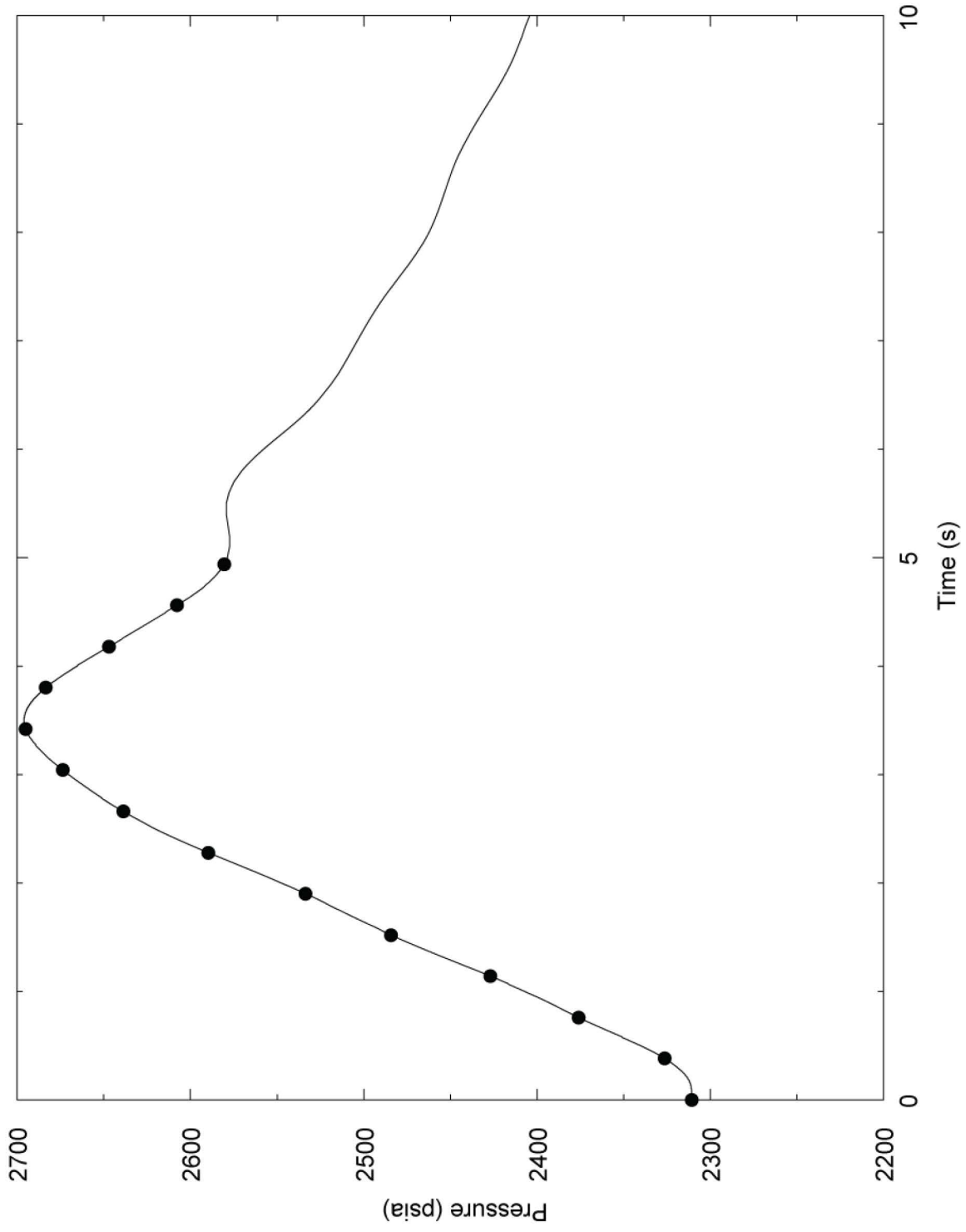
Peak Fuel Centerline Temperature versus Time – CEA Ejection (BOC HFP)



Amendment No. 26 (11/13)

<p>FLORIDA POWER AND LIGHT COMPANY ST.LUCIE PLANT UNIT 1 PEAK FUEL CENTERLINE TEMPERATURE VS TIME CEA EJECTION (BOC HFP)</p>
<p>FIGURE 15.4.5-6</p>

Peak RCS Pressure versus Time – CEA Ejection (BOC HFP)



Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1
PEAK RCS PRESSURE VS TIME
CEA EJECTION (BOC HFP)**

FIGURE 15.4.5-7

15.4.6 STEAM LINE BREAK ACCIDENT

15.4.6.1 Identification of Causes

The steam line break event is initiated by a postulated break in a main steam line. If the break is located upstream of a main steam check valve, steam flows to the break from only the upstream steam generator (because the check valve precludes backflow to the break from the other steam generator). The steam flow from that upstream steam generator increases because of the break, but the steam flow from the other steam generator may also increase to compensate for the reduced steam flow to the turbine from the upstream steam generator. If, on the other hand, the break is located downstream of the main steam check valves, the steam flows from both steam generators increase.

The increased steam flows depressurize the steam generators. The resultant primary coolant cooldown, in conjunction with a negative MTC, causes the reactor power to increase. A reactor trip occurs due to high indicated power, asymmetric steam generator pressure, or high containment pressure, thus, terminating the power excursion. The increased reactor power and, if the cooldown is asymmetric, the augmented radial power peaking reduce the margin to DNB and fuel centerline melt. The margin to DNB may further be reduced by an RCS flow coastdown triggered by a postulated loss of offsite power when the reactor scram occurs.

After reactor scram, the affected SG pressure and temperature will continue to decrease rapidly. The drop in SG pressure will initiate a main steam isolation signal (MSIS). Following appropriate delays, the main steam isolation valves (MSIVs) on both the affected and unaffected SGs will close and terminate the blowdown from the unaffected SG.

The cooldown of the RCS will insert positive reactivity from both moderator and fuel temperature reactivity feedbacks (particularly at EOC conditions with the most-negative MTC). The magnitude of core subcriticality depends on the scram worth and the moderator and fuel temperature reactivity feedbacks.

With the most reactive control rod assumed to be stuck out of the core, the radial neutron flux (and, therefore, power) distribution will be highly peaked in the region of the stuck control rod. The consequences would be most limiting if the core sector with the stuck control rod is also the sector being cooled primarily with coolant delivered to the cold leg of the affected loop.

The event will be terminated by the injection of boron from high pressure safety injection pumps and/or by the dryout of the affected steam generator which will stop the RCS cooldown.

No credit is taken for automatic isolation of auxiliary feedwater (AFW) on SG differential pressure, and AFW flow is assumed to be isolated at 10 minutes by operator action (thereby reducing the primary-to-secondary heat transfer to the affected steam generator).

15.4.6.2 Analysis of Effects and Consequences

This event has two distinct phases, i.e., prior to reactor scram or "pre-scram" and after reactor scram or "post-scram". Each of these phases was analyzed to assess the impact to fuel failure for the EPU. Detailed analyses were performed with approved methodologies using the S-RELAP5 and XCOBRA-IIIC codes (References 117 and 108). The S-RELAP5 code was used to model the key system components and calculated neutron power, fuel thermal response, surface heat transport, and fluid conditions (such as coolant flow rates, temperatures, and pressures) and produce an estimated time of MDNBR. The core fluid boundary conditions and average rod surface heat flux were then input to the XCOBRA-IIIC code, which was used to calculate the MDNBR using the HTP CHF correlation (Reference 110) for the pre-scram cases and the Biasi correlation (Reference 66) for the post-scram cases. The PRISM code was used to calculate power distribution information and kinetics parameters.

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This event is classified as a Postulated Accident, which is not expected to occur during the life of the plant, but is evaluated to demonstrate the adequacy of the design. The principally challenged acceptance criterion for this event is with respect to radiological consequences. The transient analysis documented the extent of fuel failure and calculated steam releases for input to the radiological dose analyses based on a plant cooldown to 212°F.

15.4.6.2.1 Pre-Scram Steam Line Break

The Pre-Scram steam line break event was analyzed from HFP conditions to assess the potential amount of fuel failure due to DNB and fuel melting. A full range of break sizes, up to the double-ended guillotine break of a main steam line, was considered in the analysis with the following break locations:

- Break located inside containment and upstream of an MSIV
- Break located outside containment and upstream of an MSIV
- Break located downstream of an MSIV

Also, a bounding range of negative MTC values was considered including the most negative TS limit of -32 pcm/°F. From the calculations, the most limiting combination of break size and MTC was determined based on the amount of fuel failure. Loss of offsite power was assumed to occur coincident with reactor scram in order to produce a conservative MDNBR.

The parameters and equipment states were chosen to provide conservative calculation of fuel failures. The biasing and assumptions for key input parameters were consistent with the approved methodology (Reference 117).

- Initial Conditions - For the pre-scram analysis, the event was initiated from rated power plus uncertainty conditions with a maximum core inlet temperature and minimum TS RCS flow. This set of conditions minimizes the initial margin to DNB. Loss of offsite power was assumed at the time of turbine trip resulting in the coastdown of the reactor coolant pumps.
- Break Size and Location - For the pre-scram analysis, a full range of break sizes, up to a full guillotine break of a main steam line, was considered to determine the most limiting combination of break size and MTC based on the amount of fuel failure. Breaks were modeled both upstream and downstream of the MSIVs.
- Break Flow - The Moody critical flow model was used at the SG integral flow restrictor and was modeled to maximize break flow and rate of cooldown. Steam-only flow out the break was also assumed to maximize the secondary and RCS cooldown rate.

- Reactivity Feedback - This event is primarily driven by moderator feedback as a result of the cooldown of the RCS. For the pre-scrum analysis, a bounding range of negative MTC values was considered including the most negative TS limit of -32 pcm/°F. Minimum scram worth, with the most reactive rod stuck out of the core, was assumed.
- Reactor Protection System Trips and Delays - Reactor protection trip setpoints and delay times were biased to conservatively delay the initiation of scram. In addition, harsh containment conditions were considered such that only those trips qualified for harsh environments were credited and increased uncertainties were included in the setpoints of the credited environmentally qualified trips. The analysis included the effect of power decalibration for both the NI-power (due to excore detector decalibration as a result of the severe overcooling of the fluid in the reactor vessel downcomer) and ΔT -power (due to lagged temperature signals).
- Gap Conductance - Gap conductance was set to a conservative EOC value to maximize the heat flux through the cladding and minimize the negative reactivity inserted due to Doppler feedback.
- Steam Generator Tube Plugging - No steam generator tube plugging was assumed so as to maximize the primary-to-secondary side heat transfer which exacerbates the reactivity insertion due to moderator feedback.
- RCS Flow - Cooldown of the RCPs was assumed to occur due to loss of offsite power at reactor scram.
- Single Failure - For the pre-scrum analysis, a single failure of one of the four NI detectors was assumed such that the reactor trip was conservatively delayed.

15.4.6.2.2 Post-Scrum Steam Line Break

The event was analyzed from both HZP and HFP conditions to assess the potential amount of fuel failure. Offsite power available and loss of offsite power cases were considered. For cases with loss of offsite power, power is assumed to be lost at event initiation. The largest break size (DEGB) in combination with the most negative TS MTC limit of -32 pcm/°F was analyzed. Breaks located inside or outside containment were bounded because harsh conditions were assumed for all cases.

The parameters and equipment states were chosen to provide conservative calculation of fuel failures. The biasing and assumptions for key input parameters was consistent with the approved methodology (References 117 and 120).

- Initial Conditions - For the post-scrum analysis, two sets of initial conditions were considered. First, the event was assumed to initiate from rated power conditions with a maximum core inlet temperature. Rated power conditions (i.e., coolant temperatures) represent the largest potential cooldown and consequential reactivity insertion. A second set of conditions assumed that the event initiated from a hot zero power condition with the minimum allowed TS shutdown margin. For both HFP and HZP initial conditions, cases were run with and without offsite power.
- Break Size and Location - For the post-scrum analysis, a full double-ended guillotine break of a main steam line upstream of the MSIV was considered. The blowdown of the steam generator is limited by the flow area of the integral flow restrictor. This break size and location produces the largest cooldown, which increases the potential return-to-power.
- Break Flow - Moody critical flow model was used at the SG integral flow restrictor and was modeled to maximize break flow and rate of cooldown. Steam-only flow out the break was also assumed to maximize the secondary and RCS cooldown rate.

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- Reactivity Feedback - This event is primarily driven by moderator feedback as a result of the cooldown of the RCS. For the post-scrum analysis, the most negative TS limit of -32 pcm/°F was modeled. Minimum scram worth, appropriate for the assumed initial condition, was assumed. The most reactive rod was assumed to be stuck out of the core.
- Gap Conductance - Gap conductance was set to a conservative EOC value to maximize the heat flux through the cladding and minimize the negative reactivity inserted due to Doppler feedback.
- Steam Generator Tube Plugging - No steam generator tube plugging was assumed so as to maximize the primary-to-secondary side heat transfer which exacerbates the reactivity insertion due to moderator feedback.
- RCS Flow - Cases with all RCPs running (offsite power available) and with all RCPs stopped (loss of offsite power) were analyzed to evaluate the effects of RCS flow during the post-scrum phase of the event.
- AFW Flow - AFW Flow is assumed to initiate automatically at the start of the transient, as shown in Table 15.4.6-4 (see Subsection 7.2.2.1).
- Single Failure - For the post-scrum analysis, a single failure of one of the two HPSI pumps required to be operable during plant normal operation was assumed. This single failure assumption resulted in an additional delay for boron to reach the core.

15.4.6.3 Results

15.4.6.3.1 Pre-Scram

The limiting case was initiated at HFP conditions by a postulated 3.0 ft² break in a main steam line outside the reactor containment and upstream of the main steam check valve with an MTC of -20 pcm/°F.

The input parameter biasing is summarized in Table 15.4.6-1. The limiting case was calculated based on the combination of break size, break location and MTC that resulted in the lowest MDNBR and highest peak linear heat rate (LHR). The results and sequence of events of this analysis are summarized in Table 15.4.6-4. The MDNBR was calculated and compared to the 95/95 limit for the HTP DNB correlation to ensure that the fuel failures remain below the limit specified in the radiological consequences analysis. The peak LHR was calculated and compared to the fuel centerline melt limit. The calculated fuel failures meet the fuel failure limits in the radiological dose analyses.

Key system parameters illustrating the transient are presented in Figure 15.4.6-1 through Figure 15.4.6-10. Figure 15.4.6-1 shows the core power, NI power and thermal power response as a function of time. Reactor trip occurs on a variable high power (VHP) signal based on thermal power. Figure 15.4.6-2 shows core power based on rod surface heat flux. Figure 15.4.6-3 shows the pressurizer pressure and Figure 15.4.6-4 shows the pressurizer level responses. Figure 15.4.6-5 shows the hot and cold leg temperatures. Figure 15.4.6-6 shows the total RCS flow rate where RCP coastdown occurs with an assumed loss of offsite power at the time of turbine trip. Figure 15.4.6-7 through Figure 15.4.6-10 show the SG pressures, the break flow, the steam and feedwater flow rates, and the reactivity feedback, respectively.

The calculations of record for these analyses are listed in Section 15.5, Reference (8).

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15.4.6.3.2 Post-Scram

The input parameter biasing is summarized in Table 15.4.6-2. The sequences of events are summarized in Table 15.4.6-4a for the cases initiated from HZP both with and without offsite power. Table 15.4.6-4b summarizes the sequences of events for the cases initiated from HFP both with and without offsite power.

The greatest challenge to the FCM limit occurred for the case initiated from HZP with offsite power available. Key system parameters illustrating the transient for the limiting cases are presented in Figure 15.4.6-11 to 15.4.6-20. Figure 15.4.6-11 shows the break flow rates as a function of time. Figure 15.4.6-12 through Figure 15.4.6-19 show the SG pressures, the feedwater flow rates to each steam generator, the SG mass inventories, the core inlet temperatures for the affected and unaffected loops, the pressurizer liquid level, the pressurizer pressure responses, the total HPSI flow rate, and the reactivity feedback during the event, respectively. Figure 15.4.6-20 shows the total core power as well as powers for the stuck rod, affected and unaffected regions.

The greatest challenge to the departure from nucleate boiling ratio (DNBR) limit occurred for the case initiated from HZP with a loss of offsite power. Key system parameters illustrating the transient for the limiting cases are presented in Figure 15.4.6-21 to 15.4.6-30. Figure 15.4.6-21 shows the break flow rates as a function of time. Figure 15.4.6-22 through Figure 15.4.6-29 show the SG pressures, the feedwater flow rates to each SG, the SG mass inventories, the core inlet temperatures for the affected and unaffected loops, the pressurizer liquid level, the pressurizer pressure responses, the total HPSI flow rate, and the reactivity feedback during the event, respectively. Figure 15.4.6-30 shows the total core power as well as powers for the stuck rod, affected and unaffected regions.

Results are given in Table 15.4.6-4b. Statepoints were chosen at the time of maximum core power after reactor scram. The calculated MDNBRs were compared to the 95/95 limit for the Biasi CHF correlation. The peak LHRs were evaluated relative to the FCM limit on a pin-type specific basis as needed. The fuel failures were determined to be within the limits specified in the radiological consequences analysis.

The calculation of record for this analysis is in Section 15.5, Reference (8).

EC300419

15.4.6.4 Radiological Consequences

15.4.6.4.1 Background

This event consists of a double-ended break of one main steam line either inside or outside of containment. Allowable fuel failure rates due to DNB and fuel centerline melt are determined for both break locations based upon the dose limits specified in Table 6 of RG 1.183. Depending on the location of the break, the affected steam generator (SG) rapidly depressurizes and releases the initial contents of the SG to either the environment or the containment. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Plant cool down is achieved via the remaining unaffected SG. The St. Lucie Unit 1 AST dose analysis methodology is presented in Reference 107.

15.4.6.4.2 Compliance with RG 1.183 Regulatory Positions

The MSLB dose consequence analysis followed the guidance provided in RG 1.183, Appendix E, "Assumptions for Evaluating the Radiological Consequences of a PWR Main Steam Line Break Accident," as discussed below:

1. Regulatory Position 1 - The total core inventory of the radionuclide groups utilized for determining the source term for this event is based on RG 1.183, Regulatory Position 3.1, and is provided in Table 15.4.1-1e. The inventory provided in Table 15.4.1-1e is adjusted for the fraction of fuel damaged and a radial peaking factor of 1.65 is applied. The fraction of fission product inventory in the gap available for release due to DNB is consistent with Regulatory Position 3.2 and Table 3 of RG 1.183. For fuel centerline melt, the guidance provided in RG 1.183, Appendix H, Regulatory Position 1 is used to determine the release. Gap release fractions have also been increased to account for high burnup fuel rods.
2. Regulatory Position 2 - Fuel damage is assumed for this event. It was determined that the activity released from the damaged fuel will exceed that released by the two iodine spike cases; therefore, the two iodine spike cases were not analyzed.
3. Regulatory Position 3 - The activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.
4. Regulatory Position 4 - Iodine releases from the faulted SG and the unaffected SG to the environment (or containment) are assumed to be 97% elemental and 3% organic. These fractions apply as a result of fuel damage.
5. Regulatory Position 5.1 - The primary-to-secondary leak rate is apportioned between the SGs as specified by Technical Specification 5.5.6 (0.5 gpm total, 0.25 gpm to any one SG). Thus, the tube leakage is apportioned equally between the two SGs.
6. Regulatory Position 5.2 - The density used in converting volumetric leak rates to mass leak rates is based upon RCS conditions, consistent with the plant design basis.
7. Regulatory Position 5.3 - The primary-to-secondary leakage is assumed to continue until after shutdown cooling has been placed in service and the temperature of the RCS is less than 212°F.

EC
299
478

8. Regulatory Position 5.4 - All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.
9. Regulatory Position 5.5.1 - In the faulted SG, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment (MSLB outside of containment) or the containment (MSLB inside of containment) with no mitigation. For the unaffected steam generator used for plant cooldown, a portion of the leakage is assumed to flash to vapor based on the thermodynamic conditions in the reactor and secondary immediately following plant trip when tube uncover is postulated. The primary-to-secondary leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence.
10. Regulatory Position 5.5.2 - The postulated leakage that immediately flashes to vapor is assumed to rise through the bulk water of the SG into the steam space and is assumed to be immediately released to the environment with no mitigation; i.e., no reduction for scrubbing within the SG bulk water is credited.
11. Regulatory Position 5.5.3 - All leakage that does not immediately flash is assumed to mix with the bulk water.
12. Regulatory Position 5.5.4 - The radioactivity within the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 is assumed for the iodine. The retention of particulate radionuclides in the unaffected SG is limited by the moisture carryover from the SG. The same partition coefficient of 100, as used for iodine, is assumed for other particulate radionuclides. This assumption is consistent with the SG carryover rate of less than 1%. No reduction in the release is assumed from the faulted SG.
13. Regulatory Position 5.6 - Steam generator tube bundle uncover in the intact SG is postulated for up to 45 minutes following a reactor trip for St. Lucie Unit 1. During this period, the fraction of primary-to-secondary leakage which flashes to vapor is assumed to rise through the bulk water of the SG into the steam space and is assumed to be immediately released to the environment with no mitigation. The flashing fraction is based on the thermodynamic conditions in the reactor and secondary coolant. The leakage which does not flash is assumed to mix with the bulk water in the steam generator. A conservative uncover time of 60 minutes was assumed in the analysis.

15.4.6.4.3 Other Assumptions

1. The initial RCS activity is assumed to be at the Technical Specification 3.4.16 limit of 1.0 $\mu\text{Ci/gm}$ Dose Equivalent I-131 and 518.9 $\mu\text{Ci/gm}$ DE Xe-133 gross activity. The initial SG activity is assumed to be at the Technical Specification 3.7.16 limit of 0.1 $\mu\text{Ci/gm}$ Dose Equivalent I-131.
2. The steam mass release rates for the intact SG are provided in Table 15.4.6-6.
3. The RCS fluid density used to convert the primary-to-secondary leakage from a volumetric flowrate to a mass flow rate is consistent with the RCS cooldown rate applied in the generation of the secondary steam releases. The high initial cooldown rate conservatively maximizes the fluid density. The SG tube leakage mass flow rate is provided in 15.4.6-7.
4. This RCS mass is assumed to remain constant throughout the MSLB event (no change in the RCS mass is assumed as a result of the MSLB or from the safety injection system).

EC
299
478
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299
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5. For the purposes of determining the iodine concentration of the SG secondary, the mass in the unaffected SG is assumed to remain constant throughout the event. However, it is also assumed that operator action is taken to restore water level above the top of the tubes in the unaffected steam generator within a conservative time of one hour following a reactor trip.
6. All secondary releases are postulated to occur from the ADV with the most limiting atmospheric dispersion factors. Releases from containment through the SBVS are assumed to be released from the plant stack with a filter efficiency of 99% for particulates and 95% for both elemental and organic iodine. The activity that bypasses the SBVS is released unfiltered to the environment via a ground level release from containment.
7. The initial leakage rate from containment is 0.5% of the containment air per day. This leak rate is reduced by 50% after 24 hours to 0.25% /day. 9.6% of the containment leakage is assumed to bypass the SBVS filters.
8. For the MSLB inside of containment, natural deposition of the radionuclides is credited. Containment sprays are not credited.
9. For the MSLB inside of containment, containment purge is assumed coincident with the beginning of the event. As discussed in Section 6.2.5.2.2, the Hydrogen Purge system includes a demister, HEPA prefilter, two charcoal adsorber banks in series and a HEPA afterfilter. The HEPA filters are tested to meet 99.95% minimum filter efficiency. The charcoal adsorber banks are tested to meet 99% minimum filter efficiency. The analysis uses 99.5% for the HEPA filters and 98% for the charcoal filters. The Hydrogen Purge system has no automatic containment isolation valve and must be manually isolated in the event of an accident. The release fraction associated with the fuel/gap release between 30 seconds and 285 seconds when the hydrogen purge line is manually isolated is applicable.

15.4.6.4.4 Methodology

Input assumptions used in the dose consequence analysis of the MSLB are provided in Table 15.4.6-5. The postulated accident consists of two cases; one case is based upon a double-ended break of one main steam line outside of containment, and the second case is based upon a double-ended break of one main steam line inside of containment. The primary difference between these two models is the transport of the primary-to-secondary leakage through the affected steam generator. Upon a MSLB, the affected SG rapidly depressurizes. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. Plant cooldown is achieved via the remaining unaffected SG.

The analysis for both cases assumes that activity is released as reactor coolant enters the steam generators due to primary-to-secondary leakage. All noble gases associated with this leakage are assumed to be released directly to the environment. For the break outside containment, primary-to-secondary leakage into the affected steam generator is also assumed to directly enter the atmosphere. For the break inside containment, the affected steam generator leakage is released into containment. All primary-to-secondary leakage is assumed to continue until the faulted steam generator is completely isolated at 12 hours.

Primary-to-secondary tube leakage is also postulated to occur in the unaffected SG for both cases. This activity is diluted by the contents of the steam generator and released via steaming from the ADVs until the RCS is cooled to 212°F. In addition, the analysis of both cases assumes that the initial iodine activity of both SGs is released directly to the environment. The entire contents of the faulted steam generator is released immediately, while the intact steam generator release occurs during the RCS cooldown. The secondary coolant iodine concentration is assumed to be the maximum value of 0.1 $\mu\text{Ci/gm DE I-131}$ permitted by TS. These release assumptions are consistent with the requirements of RG 1.183.

Allowable levels of fuel failure for DNB and fuel centerline melt are determined for both the MSLB outside of containment and the MSLB inside of containment. These allowable fractions are based on the dose limits specified in Table 6 of RG 1.183. The activity released from the fuel that is assumed to experience DNB is based on Regulatory Positions 3.1, 3.2, and Table 3 of RG 1.183. The activity released from the fuel that is assumed to experience fuel centerline melt is based on Regulatory Position 1 of Appendix H to RG 1.183.

For this event, the Control Room ventilation system cycles through three modes of operation:

- Initially the ventilation system is assumed to be operating in normal mode. The air flow distribution during this mode is 920 cfm of unfiltered fresh air and an assumed value of 460 cfm of unfiltered leakage.
- After the start of the event, the Control Room is isolated due to a high radiation reading in the Control Room ventilation system. A 50-second delay is applied to account for diesel generator start time, damper actuation time, instrument delay, and detector response time. After isolation, the air flow distribution consists of 0 cfm of makeup flow from the outside, 460 cfm of unfiltered leakage, and 1760 cfm of filtered recirculation flow.
- At 1.5 hours into the event, the operators are assumed to initiate makeup flow from the outside to the control room. During this operational mode, the air flow distribution consists of up to 504 cfm of filtered makeup flow, 460 cfm of unfiltered leakage, and 1256 cfm of filtered recirculation flow.
- The Control Room ventilation filter efficiencies that are applied to the filtered makeup and recirculation flows are 99% for particulate, 95% elemental iodine, and 95% organic iodine.

15.4.6.4.5 Radiological Consequences

The Control Room atmospheric dispersion factors (χ/Qs) used for this event are based on the postulated release locations and the operational mode of the control room ventilation system. The release-receptor point locations are chosen to minimize the distance from the release point to the Control Room air intake.

For the MSLB, all secondary releases are assumed to occur from the ADV that produces the most limiting χ/Qs . When the Control Room Ventilation System is in normal mode, the most limiting χ/Q corresponds to the worst air intake to the control room. When the ventilation system is isolated, the limiting χ/Q corresponds to the midpoint between the two control room air intakes. The operators are assumed to reopen the most favorable air intake at 1.5 hours. Development of control room atmospheric dispersion factors is discussed in Appendix 2J. The χ/Qs for the secondary releases are summarized in Table 15.4.6-9. For the MSLB inside of containment, the χ/Qs for containment leakage are assumed to be identical to those for the LOCA discussed in Section 15.4.1.5.4.

For the EAB dose analysis, the χ/Q factor for the zero to two-hour time interval is assumed for all time periods. Using the zero to two-hour χ/Q factor provides a more conservative determination of the EAB dose, because the χ/Q factor for this time period is higher than for any other time period. The LPZ dose is determined using the χ/Q factors for the appropriate time intervals. These χ/Q factors are provided in Appendix 2I.

The radiological consequences of the MSLB Accident are analyzed using the RADTRAD-NAI code and the inputs/assumptions previously discussed. Cases for MSLB inside and outside of containment with DNB and FCM fuel failure are analyzed. As shown in Table 15.4.6-10, the results of all four cases for EAB dose, LPZ dose, and Control Room dose are within the appropriate regulatory acceptance criteria.

The calculation of record for this analysis is in Section 15.5, Reference (8).

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15.4.6.5 Conclusion

The methodology used for analyzing steam line break events provides a conservative method of calculating the system and core responses during a MSLB event.

Fuel responses were evaluated against the DNB criterion using the Biasi CHF and HTP correlations and against the FCM criterion. Fuel failures were determined to be within the allowable limits.

EC292761

The total primary-to-secondary leakage used in the analysis is 1.0 gpm. The primary-to-secondary leakage limit in the Technical Specification has been reduced from 1 gpm total at accident conditions to 150 gallons per day through any one steam generator (SG) at room temperature. This change provides margin to account for a potential increase in leakage due to higher differential pressure under accident conditions.

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EC292761

Off-site location and control room doses have been evaluated using Regulatory Guide 1.183 acceptance criteria to establish fuel failure limits following a steam system pipe break event. For a main steam pipe rupture outside of containment, the allowable fuel failure limits are 1.2% for DNB and 0.29% for fuel centerline melt (FCM). For a pipe break inside of containment, the allowable fuel failure limits are 21% for DNB and 4.5% for FCM.

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Table 15.4.6-1

Pre-Scram Main Steam Line Break: Input Parameter Biasing

PARAMETER	PRE-SCRAM
	HFP
Reactor Power	3029.06 MWt
Average Core Inlet Temperature	551°F
Reactor Coolant Flow Rate	375,000 gpm
Pressurizer Pressure	2250 psia
Break Size	Ranged from 1.0 ft ² to the values below Inside containment: 6.3552 ft ² Outside containment, upstream of MSIV: 5.4119 ft ² Outside containment, downstream of MSIV: 6.874 ft ²
Break Location	Inside containment Outside containment, upstream of MSIV Outside containment, downstream of MSIV
Bypass Flow Rates	4.2%
Reactivity Feedback	Assume the most reactive rod is stuck out of the core
Moderator Density Reactivity	-8 pcm/°F to -32.0 pcm/°F
Doppler Reactivity Feedback	EOC
Variable High Power RPS setpoint	Indicated power 112% of rated power
Single Failure	One of two ex-core detectors next to affected sector assumed to fail
Ex-Core Detector decalibration factor	0.73% of rated power change in Indicated power/°F change in downcomer fluid temperature
RTD lag time constant	8.0 sec (cold leg) 0.0 sec (hot leg)
RPS and ESFAS trips	Harsh conditions assumed for inside-containment breaks
Scram Worth	6732.67 pcm
Feedwater Systems	Maximum MFW allowed to 125% of nominal HFP flow
AFAS	AFW was not operational
HPSI Temperature	36°F
HPSI Sweepout	165.75 ft ³
HPSI Boron Concentration	1900 ppm
TM/LP Trip	0.9 sec delay plus 0.5 sec scram rod delay
SG Tube Plugging	0%
Containment conditions	13.45 psia 2.6366E6 ft ³

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Table 15.4.6-2

Post-Steam Line Break: Input Parameter Biasing

Parameter	Post-Scram	
	HZP	HFP
Reactor Power	1W	3020 MWt
Average Core Inlet Temperature	533.3°F	551°F
Reactor Coolant Flow Rate	375,000 gpm	375,000 gpm
Pressurizer Pressure	2250 psia	2250 psia
Break Size	5.4119 ft ²	5.4119 ft ²
Bypass Flow Rates	4.2%	4.2%
Moderator Density Reactivity	-32.0 pcm/°F	-32.0 pcm/°F
Scram Worth	3600.0 pcm	6732.67 pcm
ESFAS Trips	Harsh conditions assumed	
AFAS	AFAS assumed at t=0, AFW begins after a minimum delay time	
HPSI Temperature	36°F	36°F
HPSI Sweepout	165.76 ft ³	165.76 ft ³
HPSI Boron Concentration	1900 ppm	1900 ppm
SG Tube Plugging	0%	0%
Steam Line Check Valves	Not Credited	Not Credited

Table 15.4.6-4

Pre-Scram Main Steam Line Break: Limiting Case Sequence of Events

Event	Time (sec)
Break occurs	0.0
Indicated power (ΔT signal) reached VHP trip setpoint	23.2
Reactor scram signal received and turbine tripped. Postulated loss of offsite power occurs and reactor coolant pump begins coastdown.	23.6
Maximum LHR	23.65
CEA insertion begins	24.1
MDNBR	24.85

Table 15.4.6-4a

**Post-Scram Main Steam Line Break:
HZP Sequence of Events**

Event	Offsite Power Avail. Time (sec.)	Loss of Offsite Power Time (sec.)
DEGB in main steam line occurs upstream of the MSIV, Reactor trip, AFW flow begins (both SGs)	0.0	0.0
CEA insertion begins	0.5	0.5
CEAs fully inserted	3.4	3.4
MSIS on low SG pressure	10.1	9.5
Low pressurizer pressure ESF trip	14.7	17.8
MSIVs closed	17.0	16.4
Minimum unaffected sector core inlet temperature	20.6	36.0
HPSI pumps available (unborated water begins to clear from the SI lines)	34.2	47.8
Shutdown worth has been fully overcome by moderator and Doppler feedback	81.0	171.0
Borated HPSI flow begins (unborated water has been cleared from the SI lines)	254.7	---
Peak post-scram reactor power	256.0	234.0
MDNBR	256.0	234.0
Maximum LHR	256.0	234.0
Borated HPSI flow begins (unborated water has been cleared from the SI lines)	---	263.4
Operator terminated AFW. Affected SG inventory begins to decrease.	600.0	600.0

Table 15.4.6-4b

**Post-Scram Main Steam Line Break:
HFP Sequence of Events**

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Event	Offsite Power Avail. Time (sec.)	Loss of Offsite Power Time (sec.)
DEGB in main steam line occurs upstream of the MSIV	0.0	0.00
Reactor and turbine trip	7.0	11.3
CEA insertion begins	7.5	11.8
CEAs fully inserted	10.4	14.7
MSIS on low SG pressure	16.4	11.3
MSIVs closed	23.3	18.2
Minimum unaffected sector core inlet temperature	27.0	33.4
HPSI pumps available (RCS pressure higher than the HPSI pump shutoff head)	40.3	58.3
Affected SG MFW isolation valves closed	76.4	71.3
Non-borated HPSI flow begins (unborated water being cleared from the SI lines)	~100	---
AFW flow begins (both SGs)	170.0	170.0
Operator terminates AFW. Affected SG mass inventory begins to decrease.	---	600.0
Shutdown worth has been fully overcome by moderator and Doppler feedback	233.0	695.0
Operator terminates AFW. Affected SG mass inventory begins to decrease.	600.0	---
Peak post-scram reactor power	602.0	2,412.0
MDNBR	602.0	2,412.0
Maximum LHR	602.0	2,412.0

Table 15.4.6-4c

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TABLE 15.4.6-5

Main Steam Line Break (MSLB) – Inputs and Assumptions

Input/Assumption	Value
Core Power Level	3030 MW _{th} (3020 + 0.3%)
Core Average Fuel Burnup	49,000 MWD/MTU
Fuel Enrichment	1.5 – 5.0 w/o
Maximum Radial Peaking Factor	1.65
% DNB for MSLB Outside of Containment	1.2%
% DNB for MSLB Inside of Containment	21%
% Fuel Centerline Melt for MSLB Outside of Containment	0.29%
% Fuel Centerline Melt for MSLB Inside of Containment	4.5%
Core Fission Product Inventory	Table 15.4.1-1e
Initial RCS Equilibrium Activity	1.0 μCi/gm DE I-131 and 518.9 μCi/gm DE Xe-133 gross activity (Table 15.4.1-9)
Initial Secondary Side Equilibrium Iodine Activity	0.1 μCi/gm DE I-131 (Table 15.4.6-8)
Release Fraction from DNB Fuel Failures	RG 1.183, Section 3.2
Release Fraction from Centerline Melt Fuel Failures	RG 1.183, Section 3.2, and Section 1 of Appendix H
Steam Generator Tube Leakage	0.25 gpm per SG (Table 15.4.6-7)
Time to Terminate SG Tube Leakage	12.4 hours
Steam Release from Intact SGs	Table 15.4.6-6
Intact SG Tube Uncovery Following Reactor Trip Time to tube recovery Flashing Fraction	1 hour 5 %
Steam Generator Secondary Side Partition Coefficient	Unaffected SG - 100 Faulted SG - None
Time to Reach 212 °F and Terminate Steam Release	12.4 hours
Containment Volume Containment Leakage Rate 0 to 24 hours after 24 hours	2.506E+06 ft ³ 0.5% (by volume)/day 0.25% (by volume)/day
Secondary Containment Filter Efficiency	Particulate – 99% Elemental – 95% Organic – 95%
Secondary Containment Drawdown Time	310 seconds
Secondary Containment Bypass Fraction	9.6%
RCS Mass	406,715 lb _m Minimum mass used for fuel failure dose contribution to maximize SG tube leakage activity.

TABLE 15.4.6-5 (Cont'd)

Main Steam Line Break (MSLB) – Inputs and Assumptions

SG Secondary Side Mass	minimum – 120,724 lb _m (per SG) maximum – 226,800 lb _m (per SG) Maximum mass used for faulted SG to maximize secondary side dose contribution. Minimum mass used for intact SG to maximize steam release nuclide concentration.
Chemical Form of Iodine Released from SGs	Particulate – 0% Elemental – 97% Organic – 3%
Atmospheric Dispersion Factors Offsite Onsite	Appendix 2I Table 15.4.6-9 and Appendix 2J
Control Room Ventilation System Time of Control Room Ventilation System Isolation Time of Control Room Filtered Makeup Flow Control Room Unfiltered Inleakage	50 seconds 1.5 hours 460 cfm
Breathing Rates	RG 1.183 Sections 4.1.3 and 4.2.6
Control Room Occupancy Factor	RG 1.183 Section 4.2.6
Containment Natural Deposition Coefficients	Aerosols – 0.1 hr ⁻¹ Elemental Iodine – 2.89 hr ⁻¹ Organic Iodine – None

TABLE 15.4.6-6

MSLB Steam Release Rate

Time (hours)	Intact SG Steam Release Rate (lb_m/min)
0	5225
0.50	2687
2.00	2711
12.40	0.00

TABLE 15.4.6-7

MSLB Steam Generator Tube Leakage

Time (hr)	SG Tube Leakage per SG (lb_m/min)
0	1.552
0.50	1.680
0.75	1.768
1.39	1.783
2.00	1.828
4.00	1.878
6.00	1.923
8.00	1.973
10.50	2.006
12.40	0.000

TABLE 15.4.6-8

Secondary Side Source Term

Isotope	μCi/gm
I-131	8.425E-02
I-132	1.689E-02
I-133	8.713E-02
I-134	7.73E-03
I-135	3.933E-02

TABLE 15.4.6-9

Control Room χ/Q (for releases from the SGs)

Time (hours)	χ/Q (sec/m ³)
0	6.30E-03
0.01389	2.84E-03
1.5	1.62E-03
2	1.32E-03
8	5.06E-04
24	3.88E-04
96	3.30E-04
720	3.30E-04

TABLE 15.4.6-10

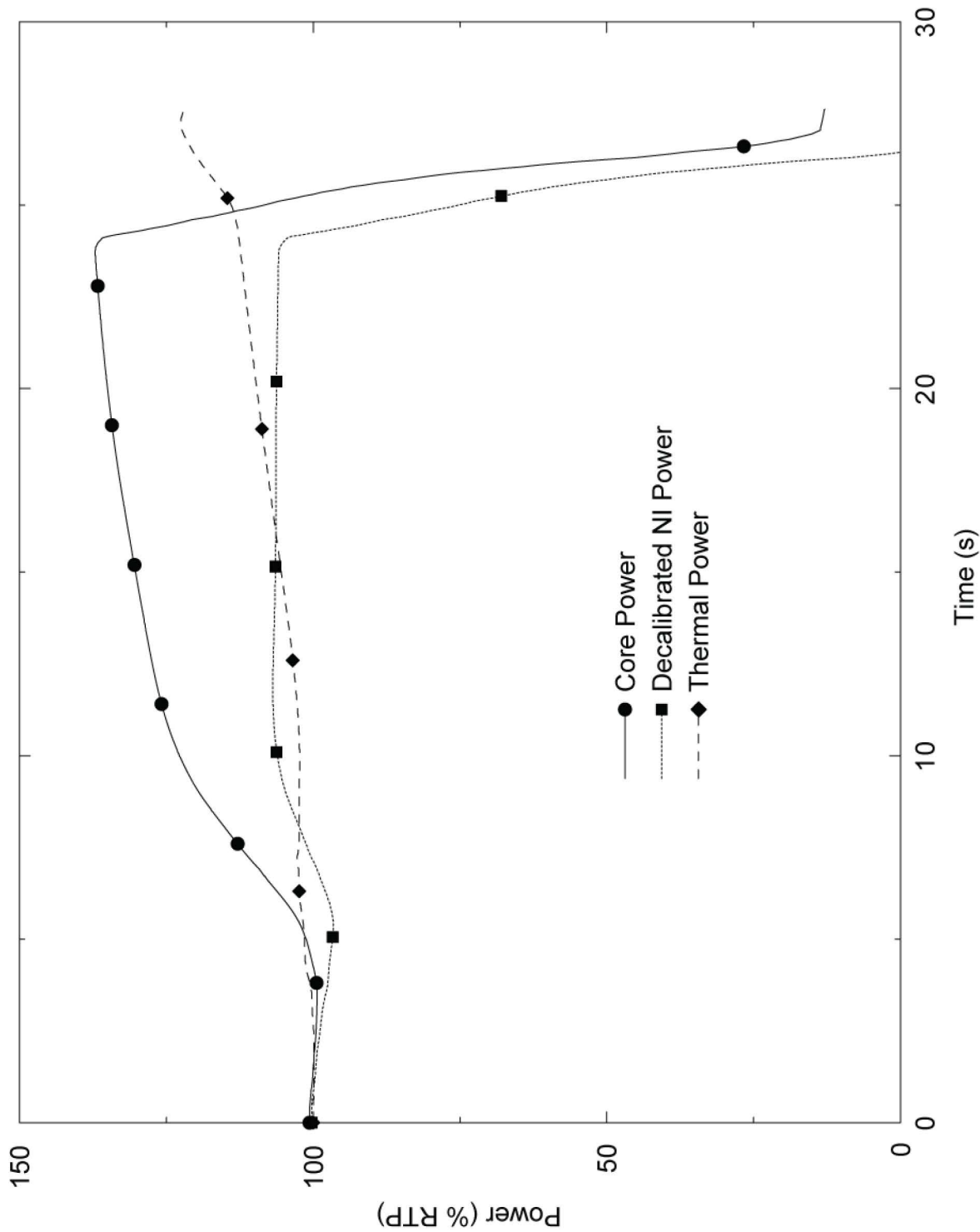
MSLB Dose Consequences

Case	Fuel Failure	EAB Dose ⁽¹⁾ (REM TEDE)	LPZ Dose ⁽²⁾ (REM TEDE)	Control Room Dose ⁽²⁾ (REM TEDE)
MSLB – Outside of Containment	1.2% DNB	0.27	0.77	4.63
MSLB – Outside of Containment	0.29% FCM	0.30	0.81	4.72
MSLB – Inside of Containment	21% DNB	0.41	0.87	4.67
MSLB – Inside of Containment	4.5% FCM	0.63	1.21	4.62
Acceptance Criteria		2.5	2.5	5

⁽¹⁾ Worst 2-hour dose

⁽²⁾ Integrated 30-day dose

Reactor Power – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)

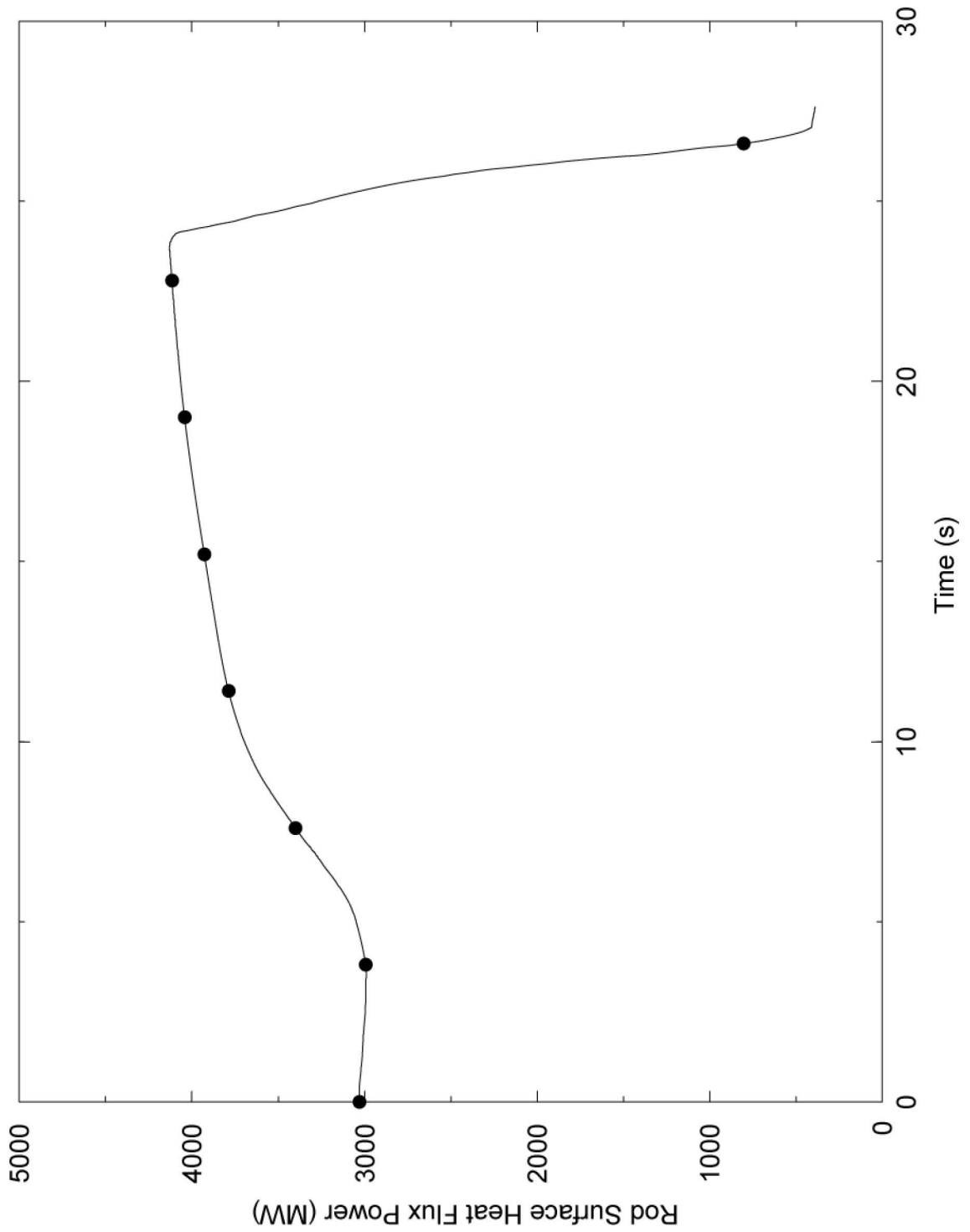


Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1
REACTOR POWER
PRE-SCRAM MAIN STEAM LINE BREAK
(HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)**

FIGURE 15.4.6-1

Total Core Heat Flux Power – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)



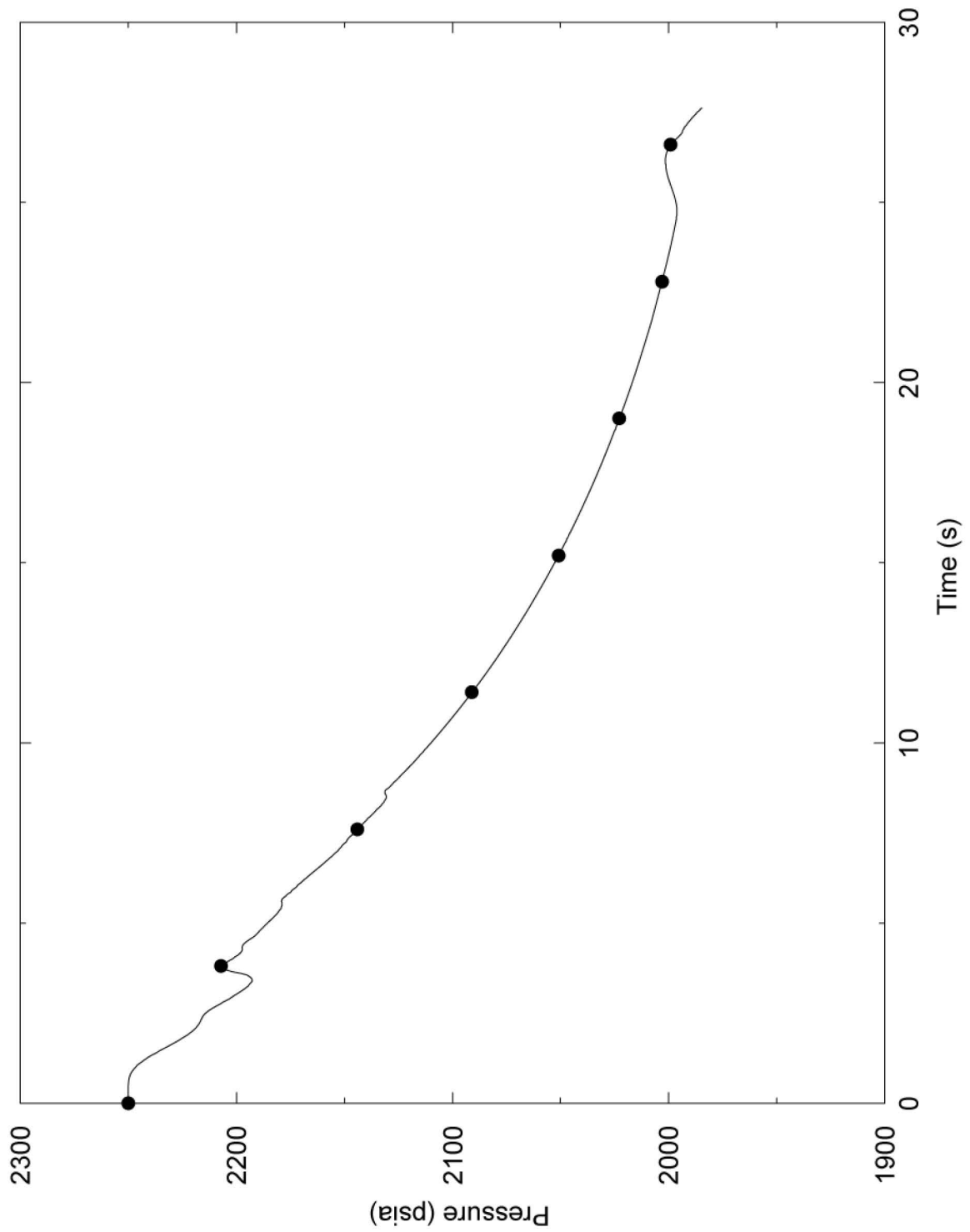
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1**

**TOTAL CORE HEAT FLUX POWER
PRE-SCRAM MAIN STEAM LINE BREAK
(HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)**

FIGURE 15.4.6-2

Pressurizer Pressure – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)



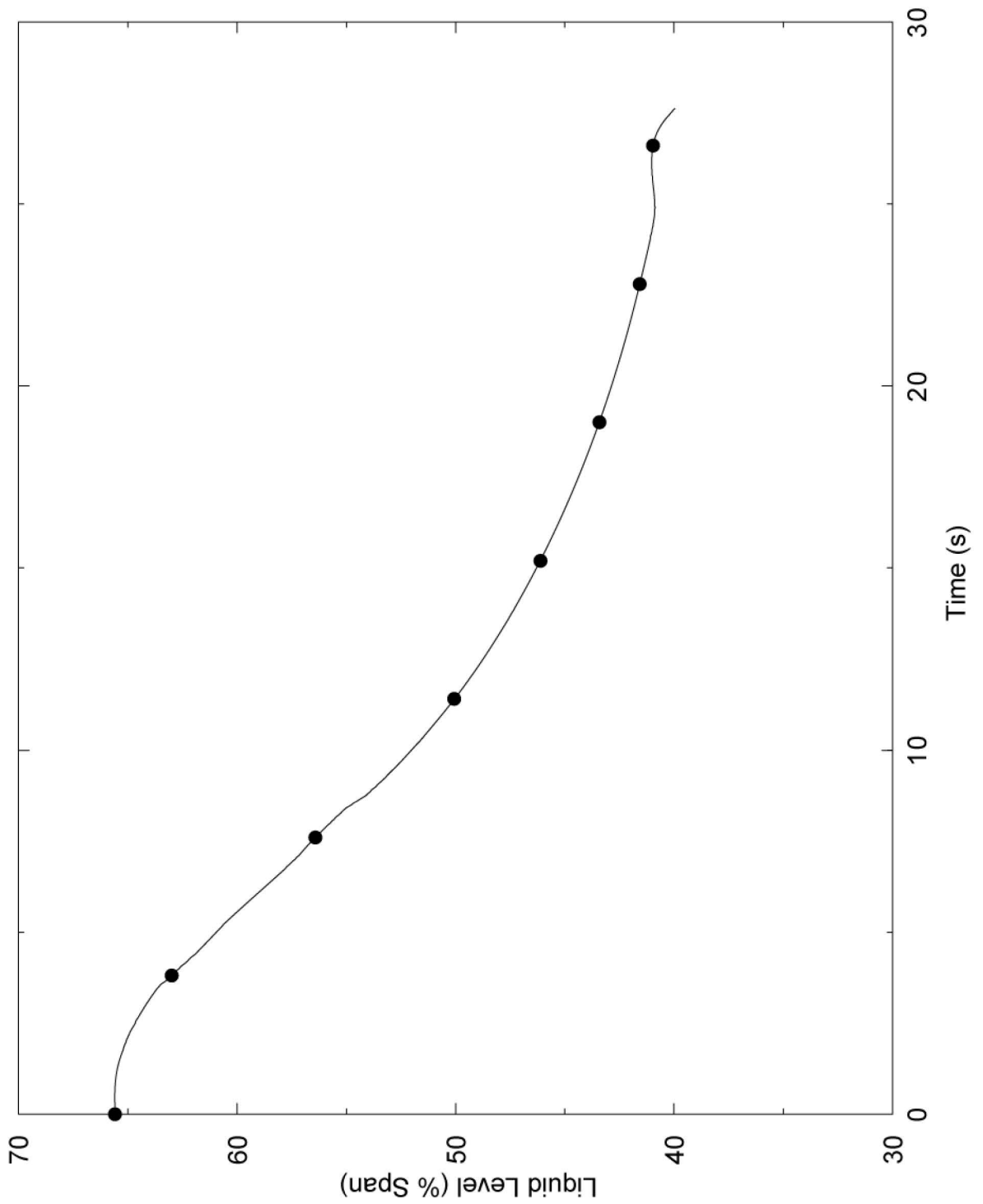
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**PRESSURIZER PRESSURE
PRE-SCRAM MAIN STEAM LINE BREAK
(HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)**

FIGURE 15.4.6-3

Pressurizer Liquid Level – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)



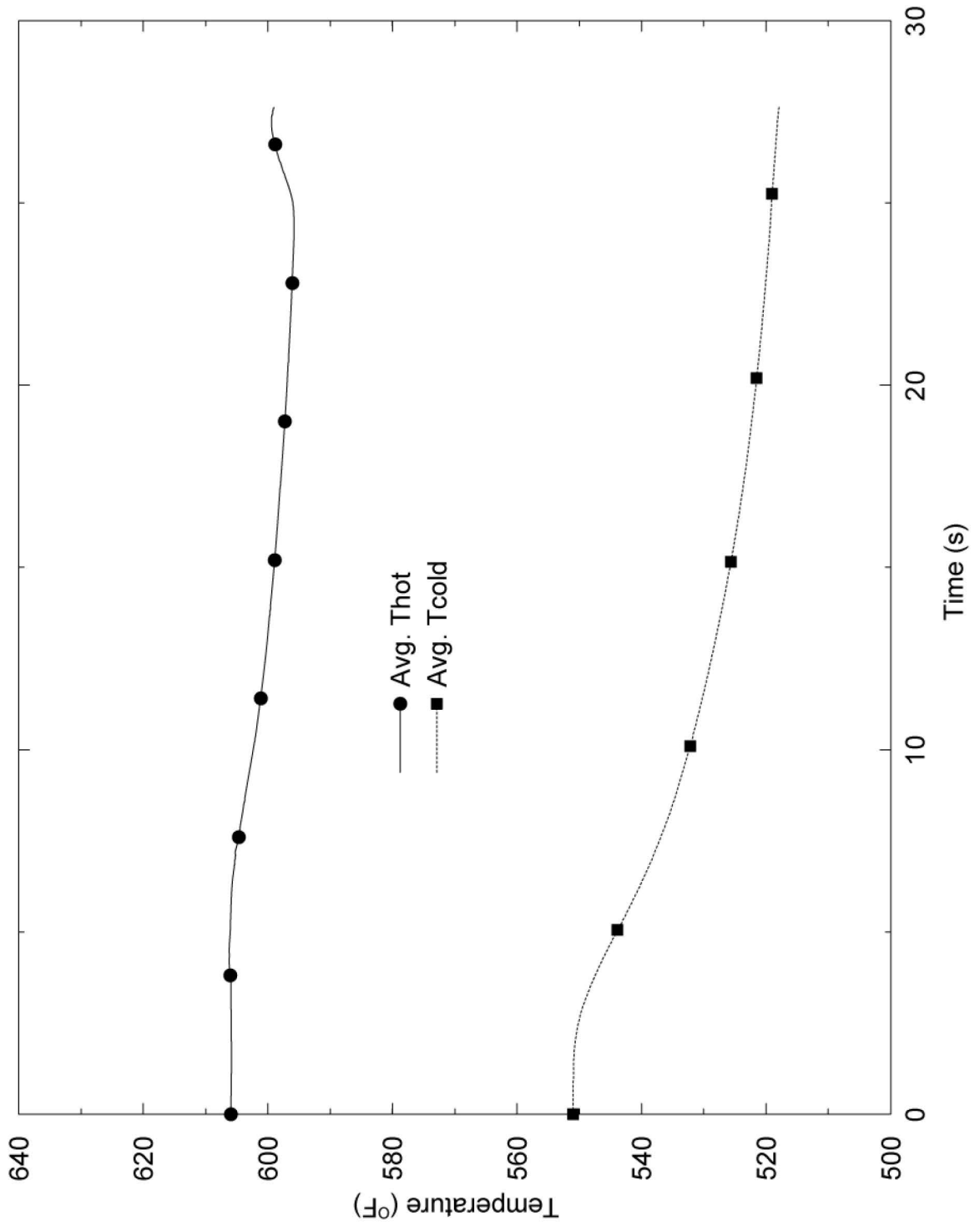
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1**

**PRESSURIZER LIQUID LEVEL
PRE-SCRAM MAIN STEAM LINE BREAK
(HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)**

FIGURE 15.4.6-4

RCS Loop Temperatures – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)



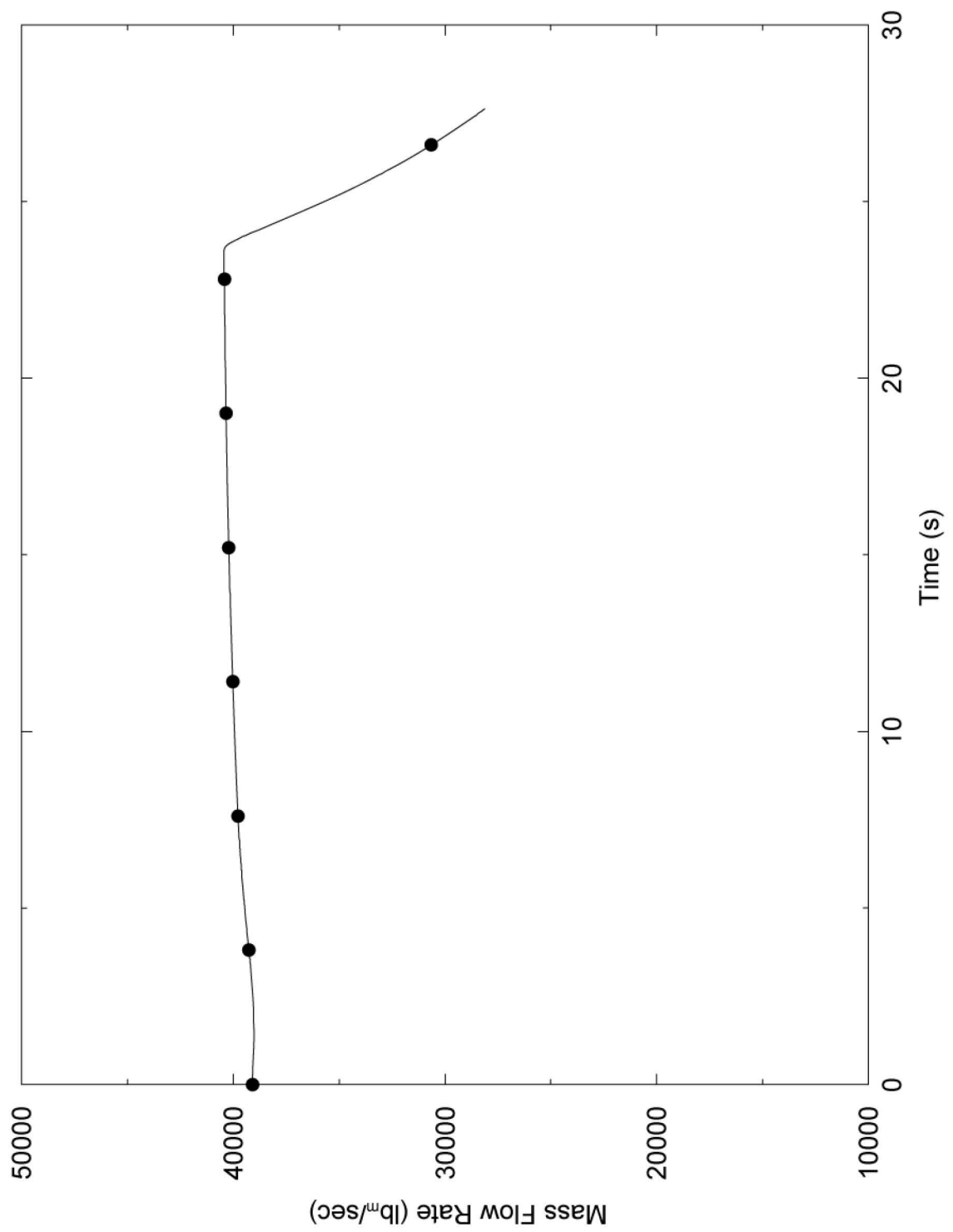
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**RCS LOOP TEMPERATURES
PRE-SCRAM MAIN STEAM LINE BREAK
(HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)**

FIGURE 15.4.6-5

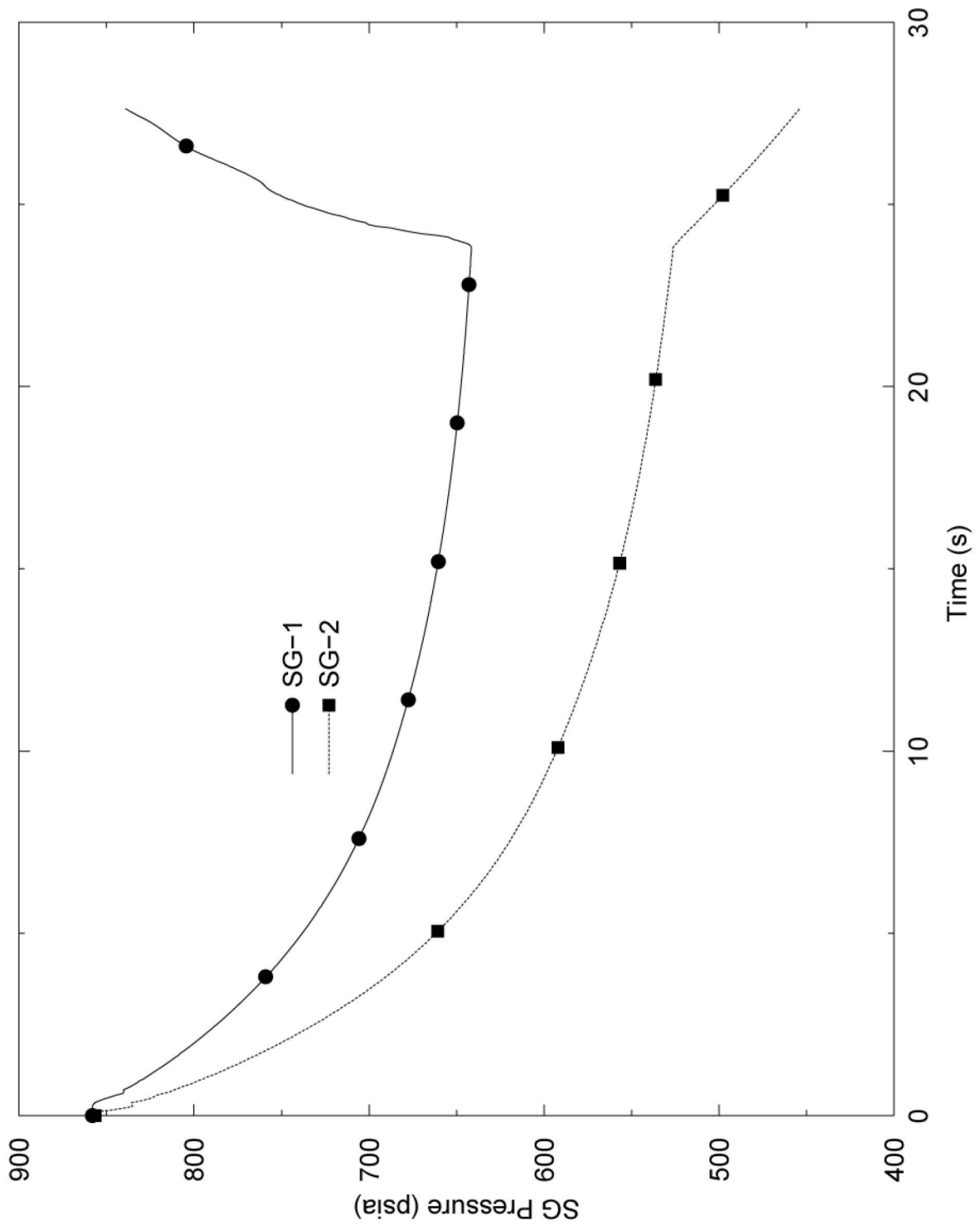
RCS Total Loop Flow Rate – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)



Amendment No. 26 (11/13)

<p>FLORIDA POWER AND LIGHT COMPANY ST.LUCIE PLANT UNIT 1</p>
<p>RCS TOTAL LOOP FLOW RATE PRE-SCRAM MAIN STEAM LINE BREAK (HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)</p>
<p>FIGURE 15.4.6-6</p>

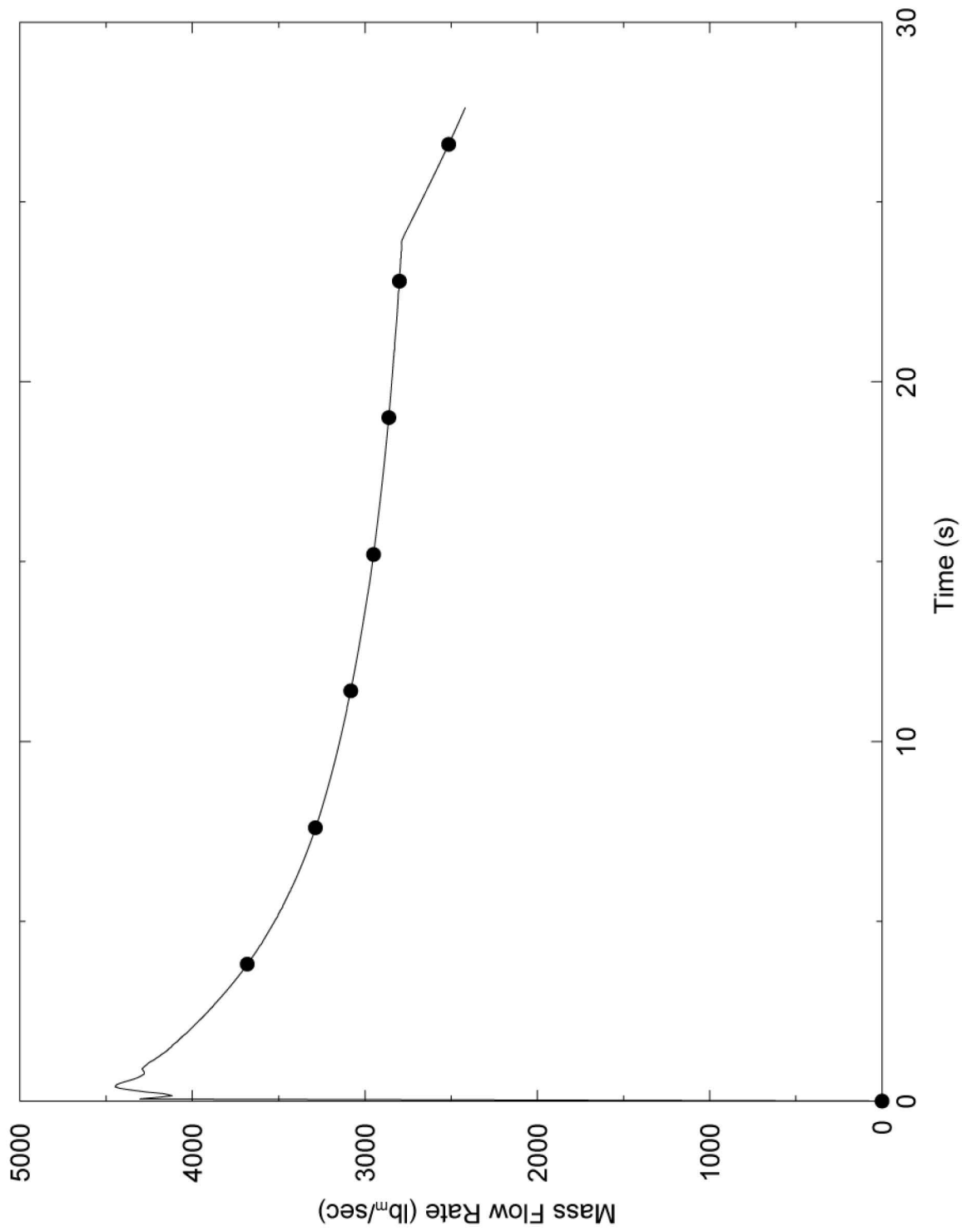
Steam Generator Pressure – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)



Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1
STEAM GENERATOR PRESSURE
PRE-SCRAM MAIN STEAM LINE BREAK
(HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)
FIGURE 15.4.6-7**

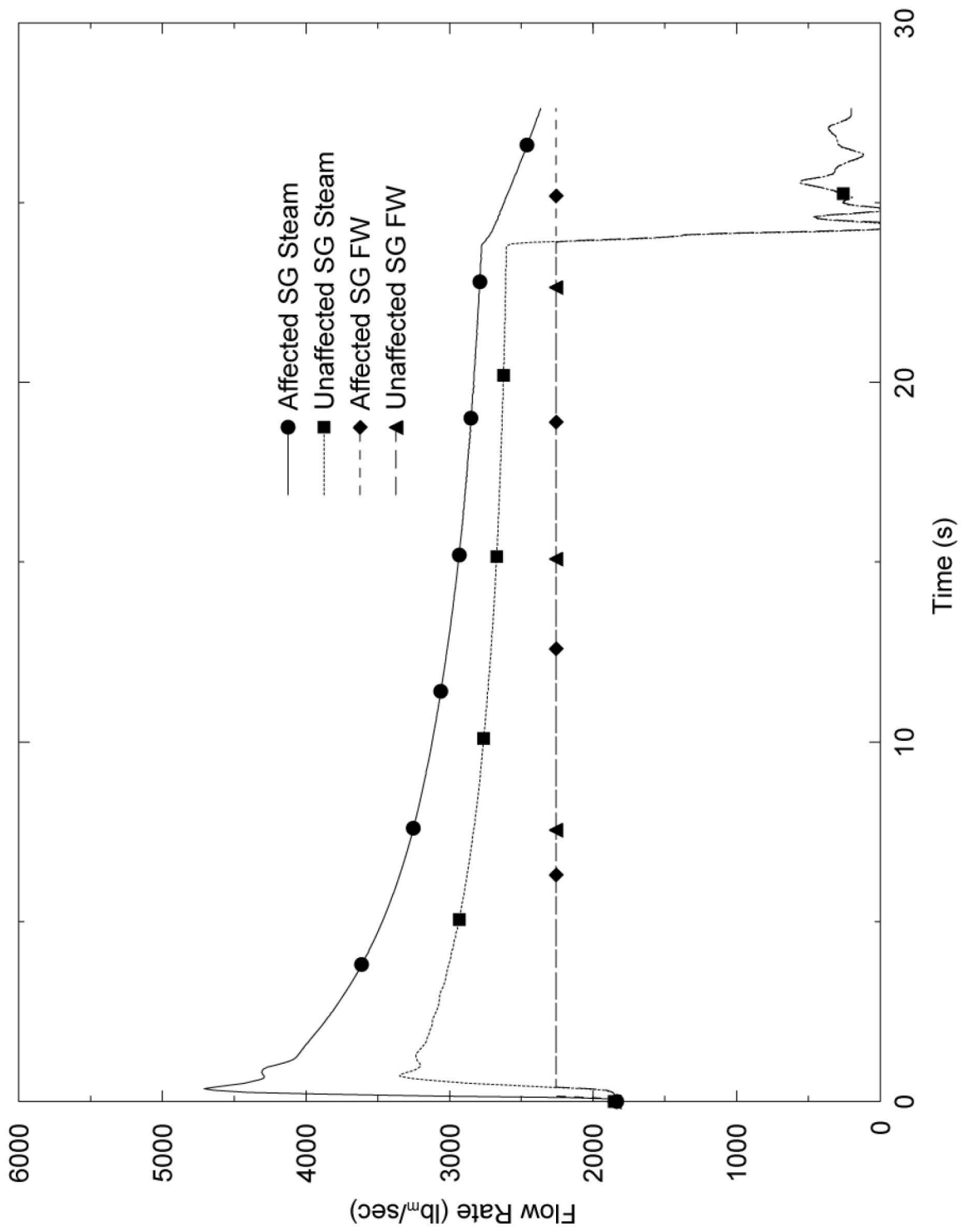
Break Flow Rate – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)



Amendment No. 26 (11/13)

<p>FLORIDA POWER AND LIGHT COMPANY ST.LUCIE PLANT UNIT 1</p>
<p>BREAK FLOW RATE PRE-SCRAM MAIN STEAM LINE BREAK (HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)</p>
<p>FIGURE 15.4.6-8</p>

Steam and Feedwater Flow Rates – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)



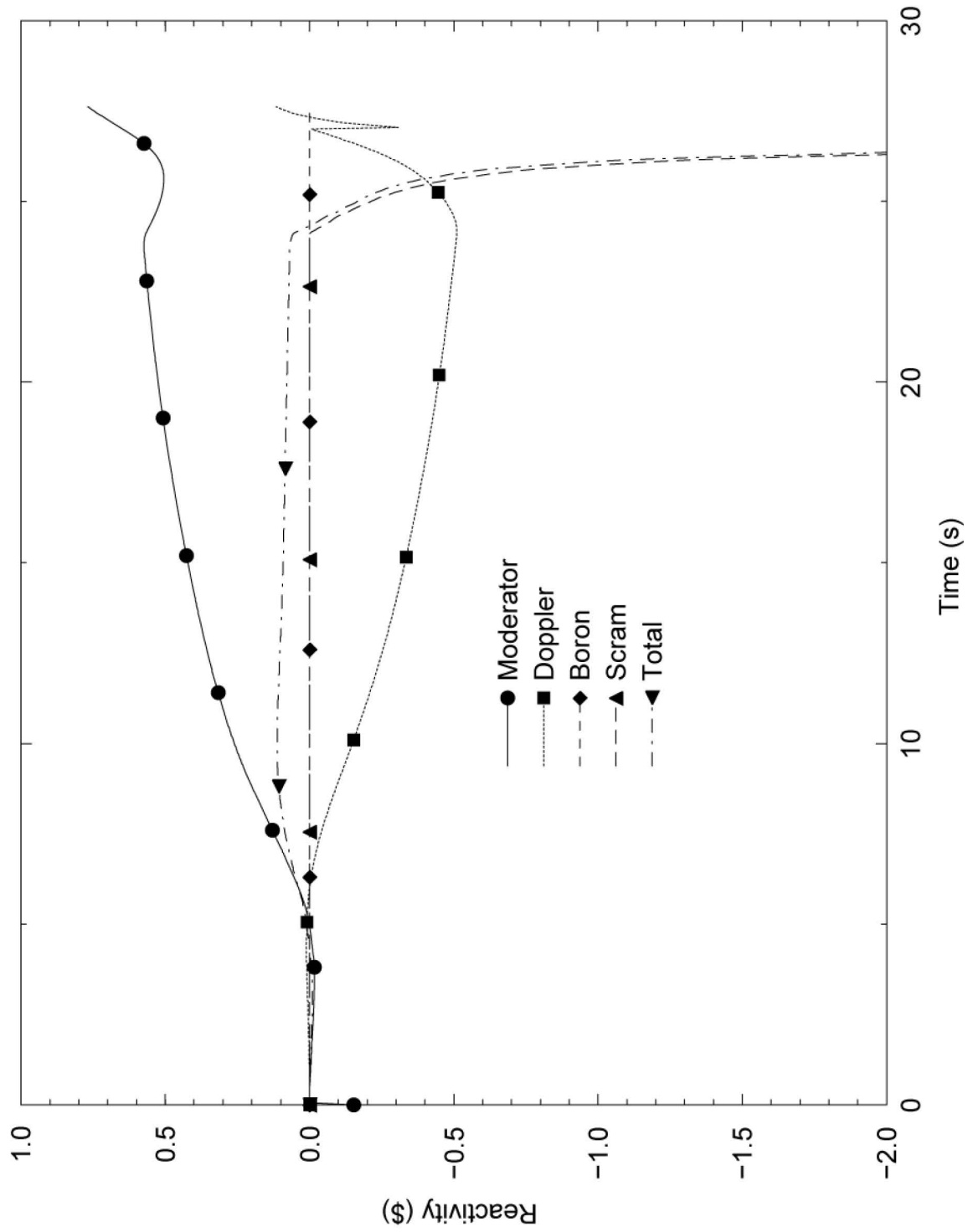
Amendment No. 26 (11/13)

FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1

STEAM AND FEEDWATER FLOW RATES
PRE-SCRAM MAIN STEAM LINE BREAK
(HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)

FIGURE 15.4.6-9

Reactivity Feedback – Pre-Scram Main Steam Line Break (HFP, 3.0 ft² break, MTC -20 pcm/°F)



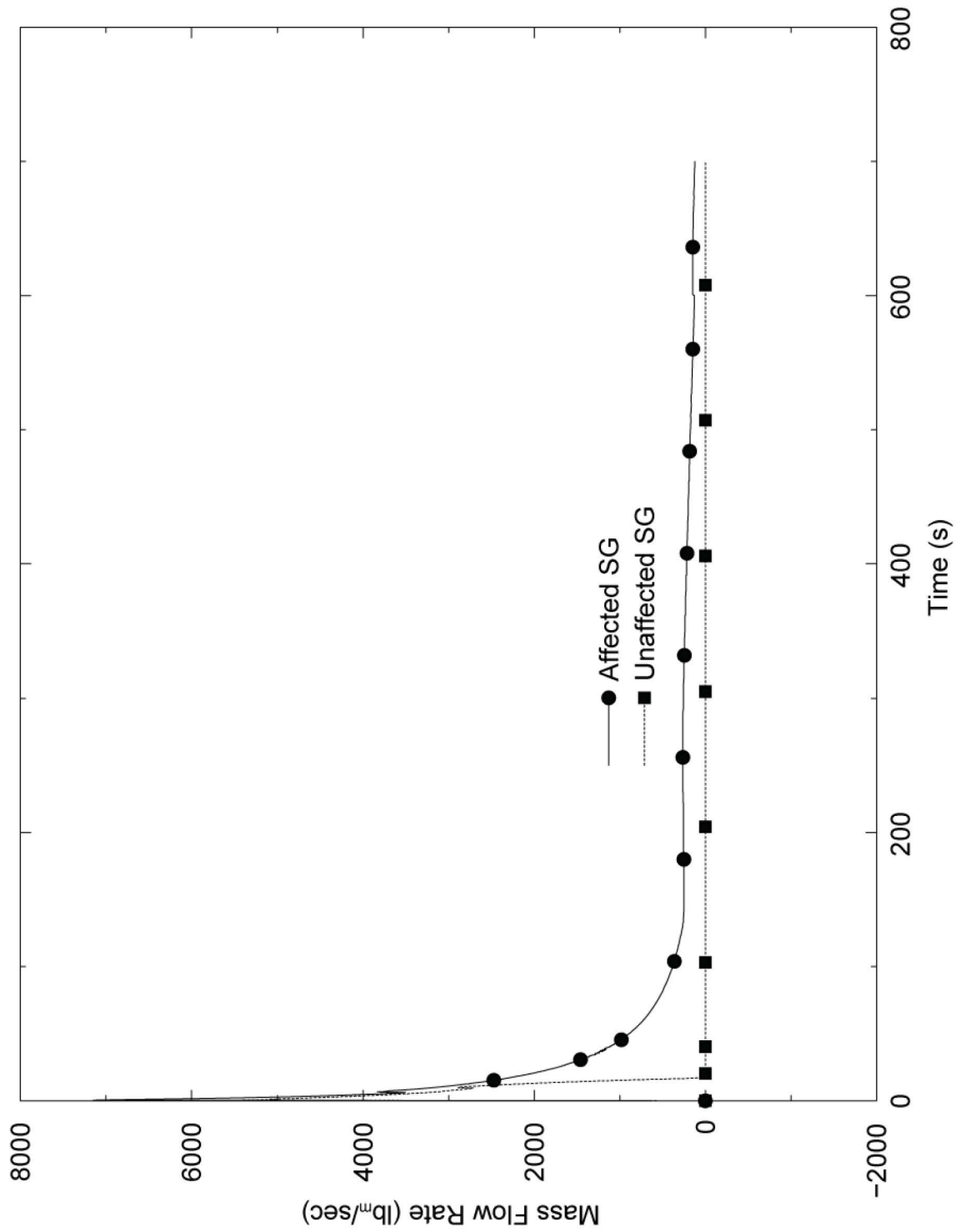
Amendment No. 26 (11/13)

FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTIVITY FEEDBACK
PRE-SCRAM MAIN STEAM LINE BREAK
(HFP, 3.0 FT² BREAK, MTC -20 PCM/°F)

FIGURE 15.4.6-10

Break Flow Rates – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)



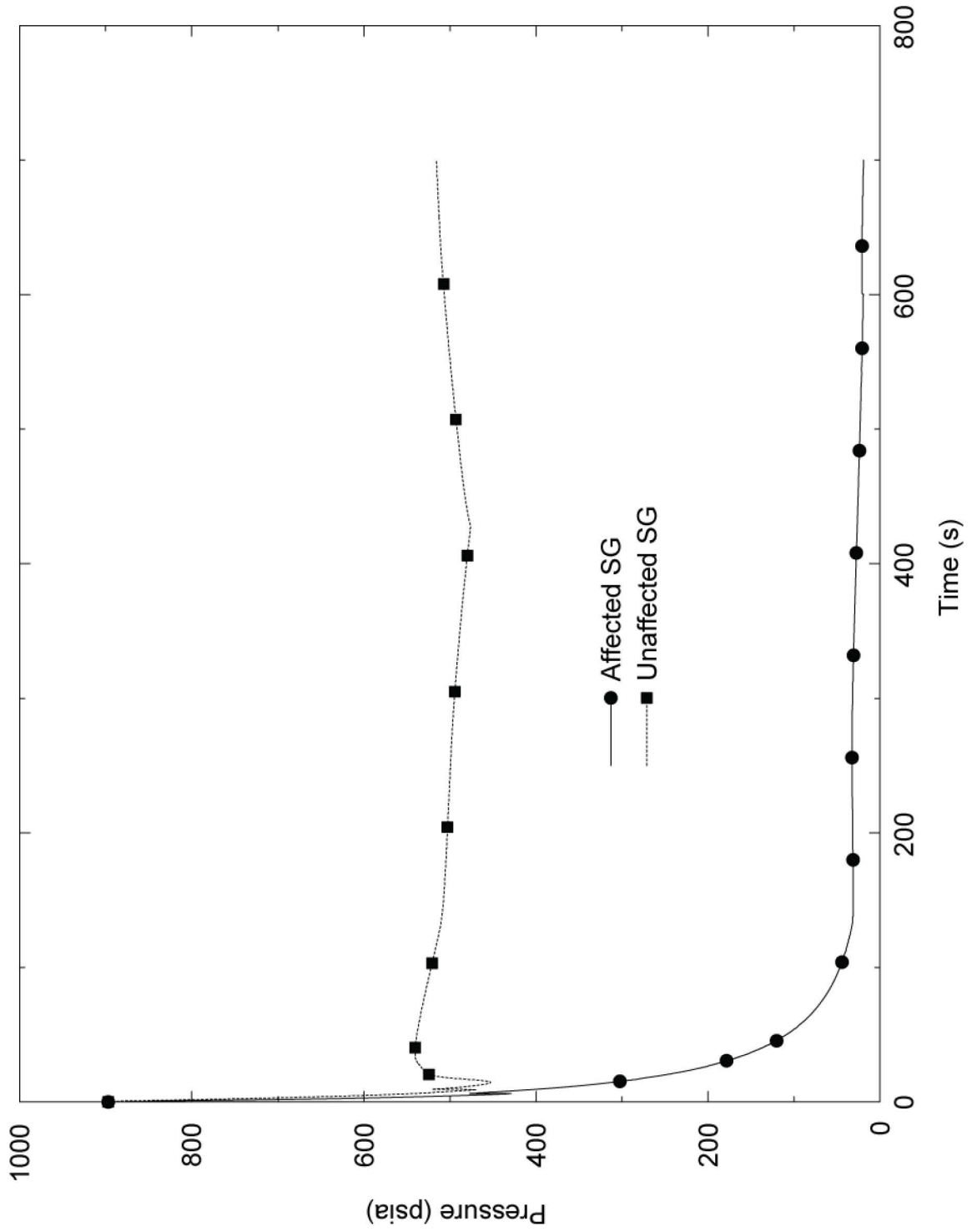
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**BREAK FLOW RATES
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, OFFSITE POWER AVAILABLE)**

FIGURE 15.4.6-11

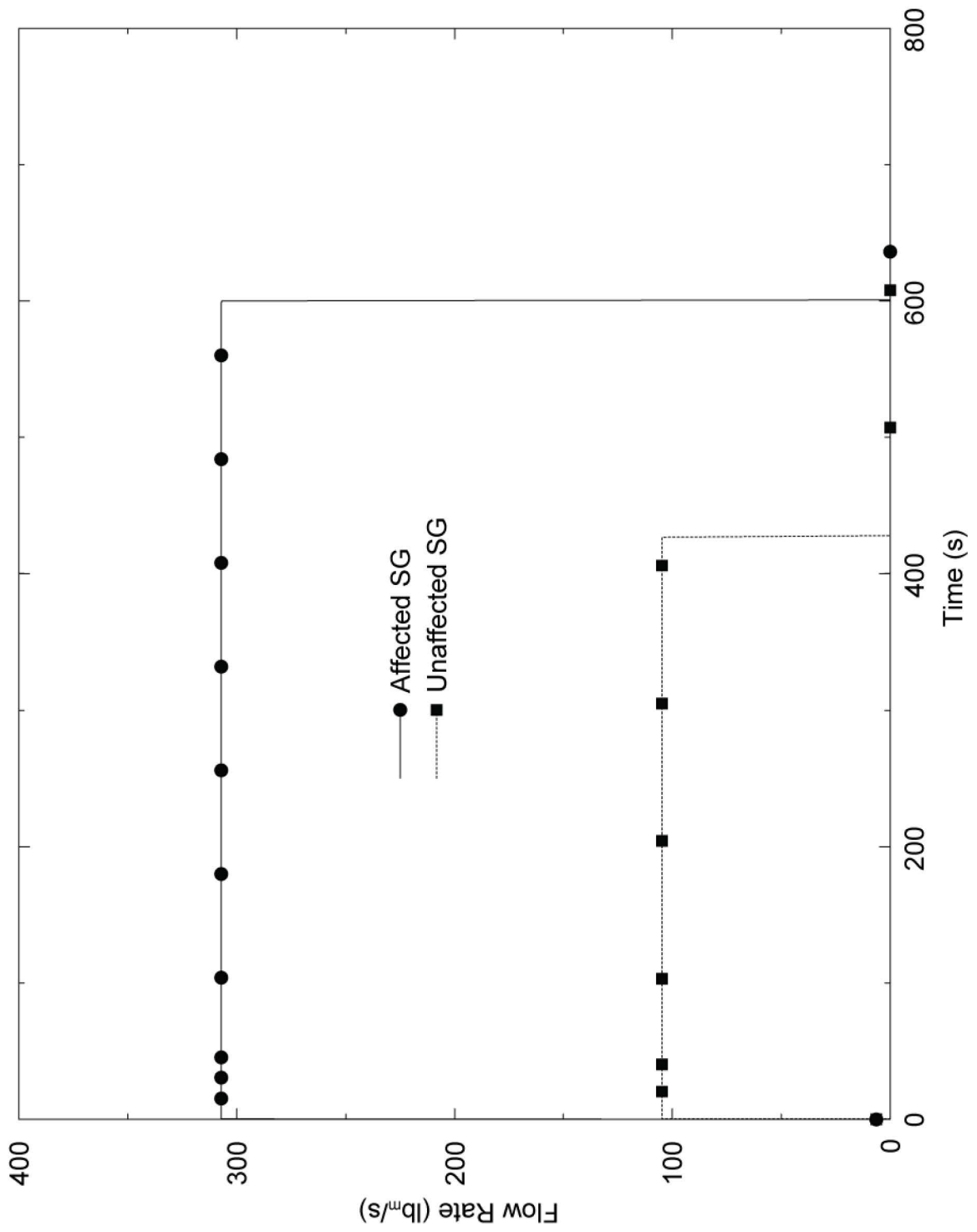
Steam Generator Pressures – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)



Amendment No. 26 (11/13)

<p>FLORIDA POWER AND LIGHT COMPANY ST.LUCIE PLANT UNIT 1</p>
<p>STEAM GENERATOR PRESSURES POST-SCRAM MAIN STEAM LINE BREAK (HZP, OFFSITE POWER AVAILABLE)</p>
<p>FIGURE 15.4.6-12</p>

Combined MFW and AFW Flow Rates – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)

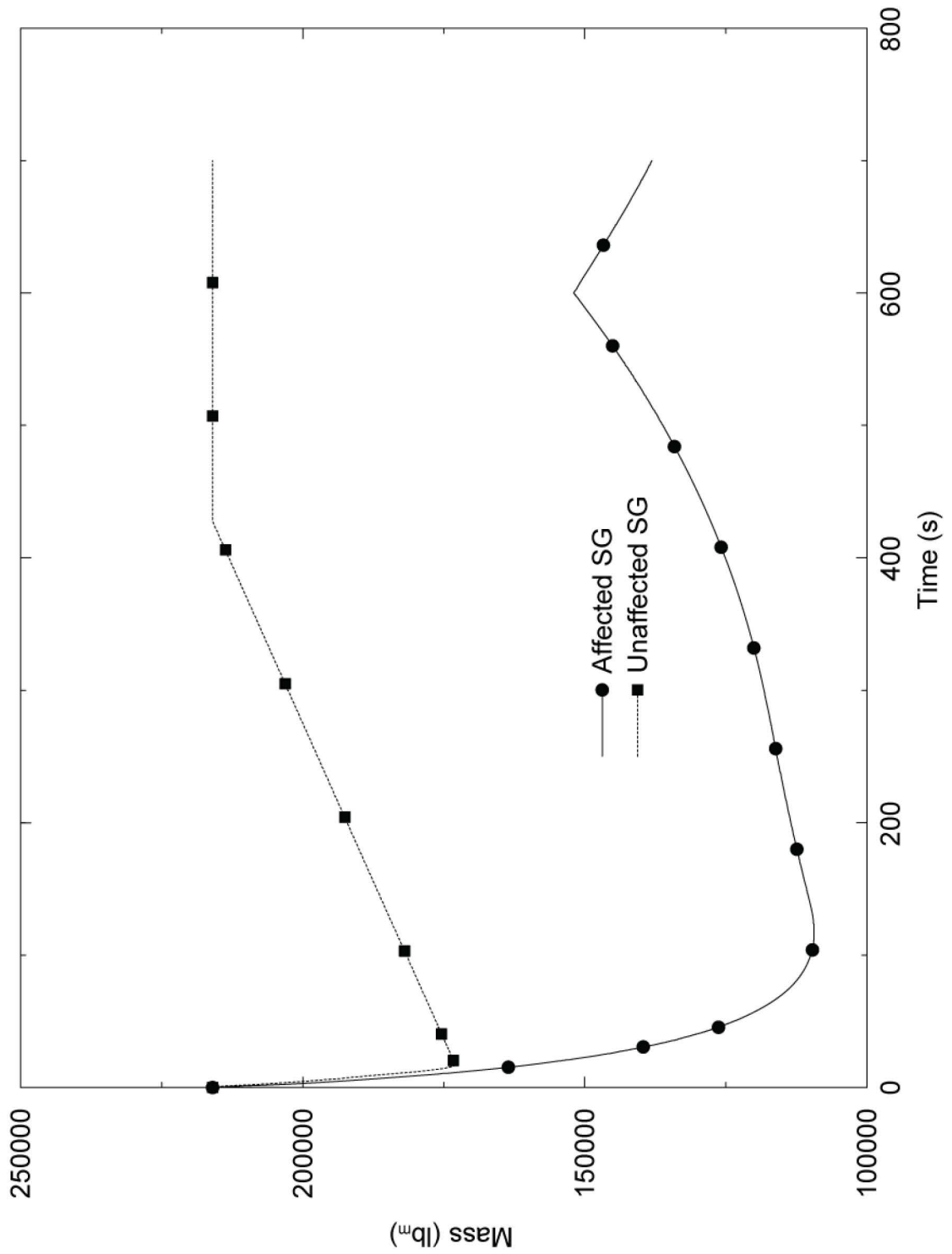


Amendment No. 26 (11/13)

FLORIDA POWER AND LIGHT COMPANY
 ST.LUCIE PLANT UNIT 1
 COMBINED MFW AND AFW FLOW RATES
 POST-SCRAM MAIN STEAM LINE BREAK
 (HZP, OFFSITE POWER AVAILABLE)

FIGURE 15.4.6-13

Steam Generator Mass Inventories – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)

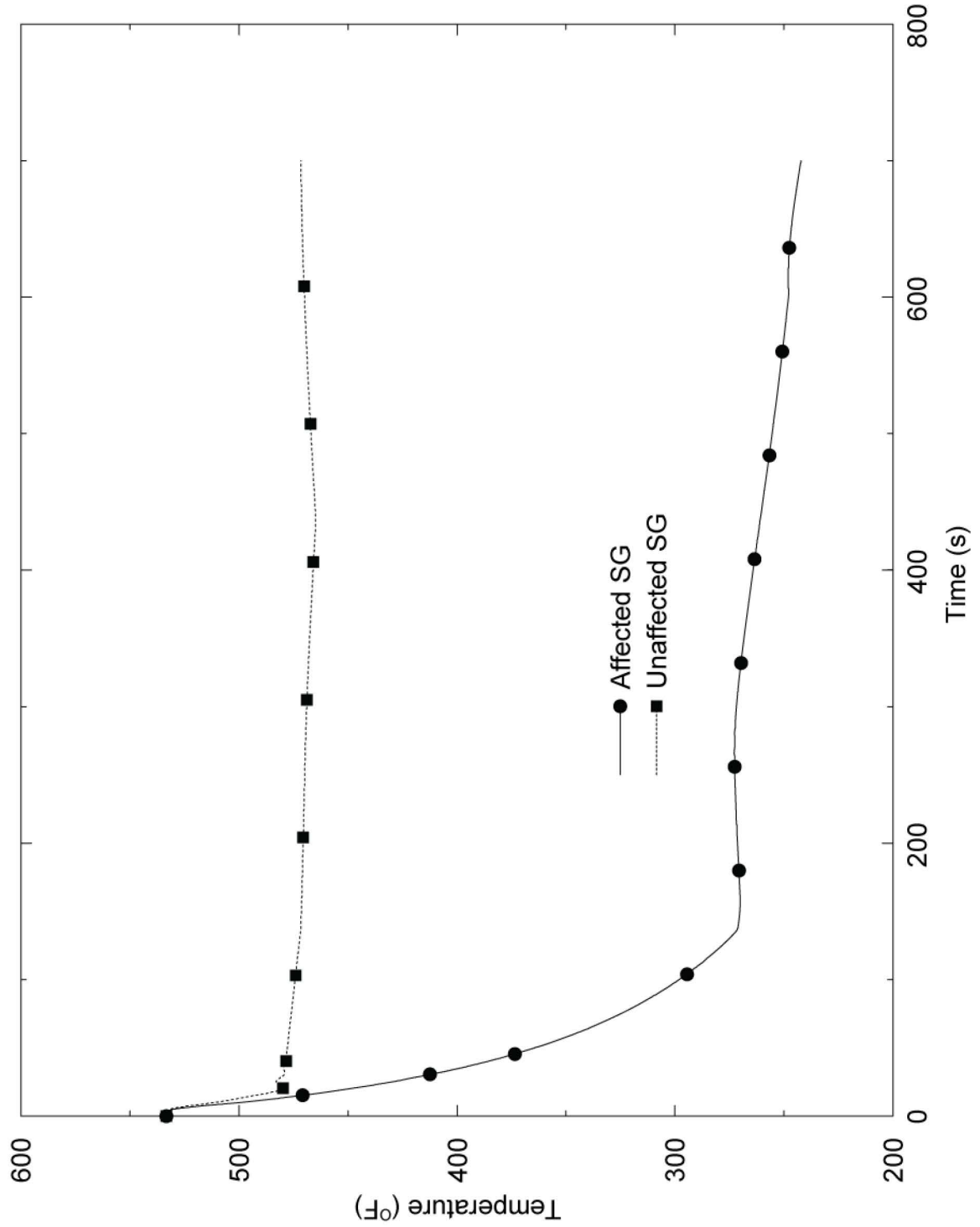


Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1
STEAM GENERATOR MASS INVENTORIES
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, OFFSITE POWER AVAILABLE)**

FIGURE 15.4.6-14

Core Inlet Fluid Temperatures – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)



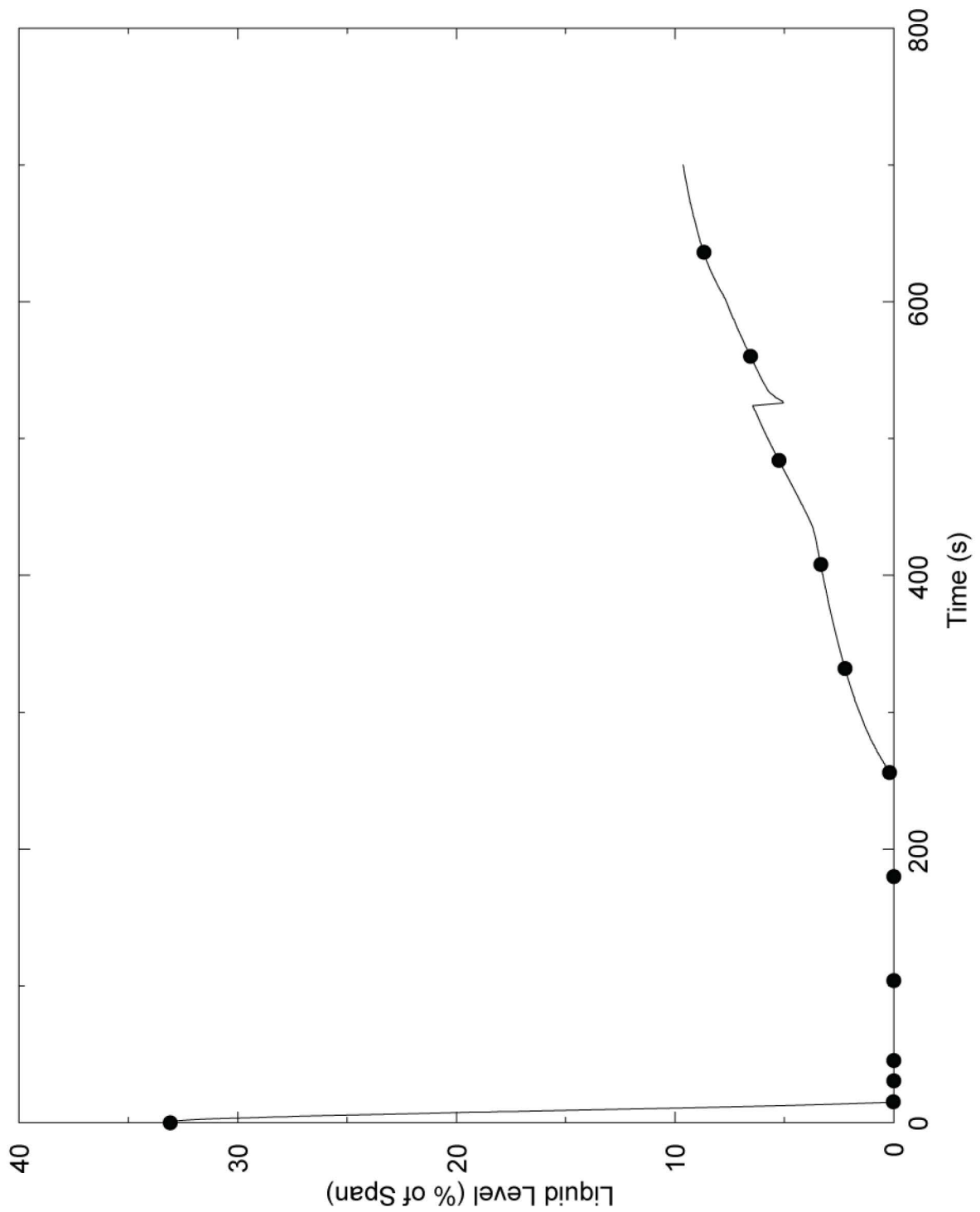
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**CORE INLET FLUID TEMPERATURES
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, OFFSITE POWER AVAILABLE)**

FIGURE 15.4.6-15

Pressurizer Liquid Level – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)



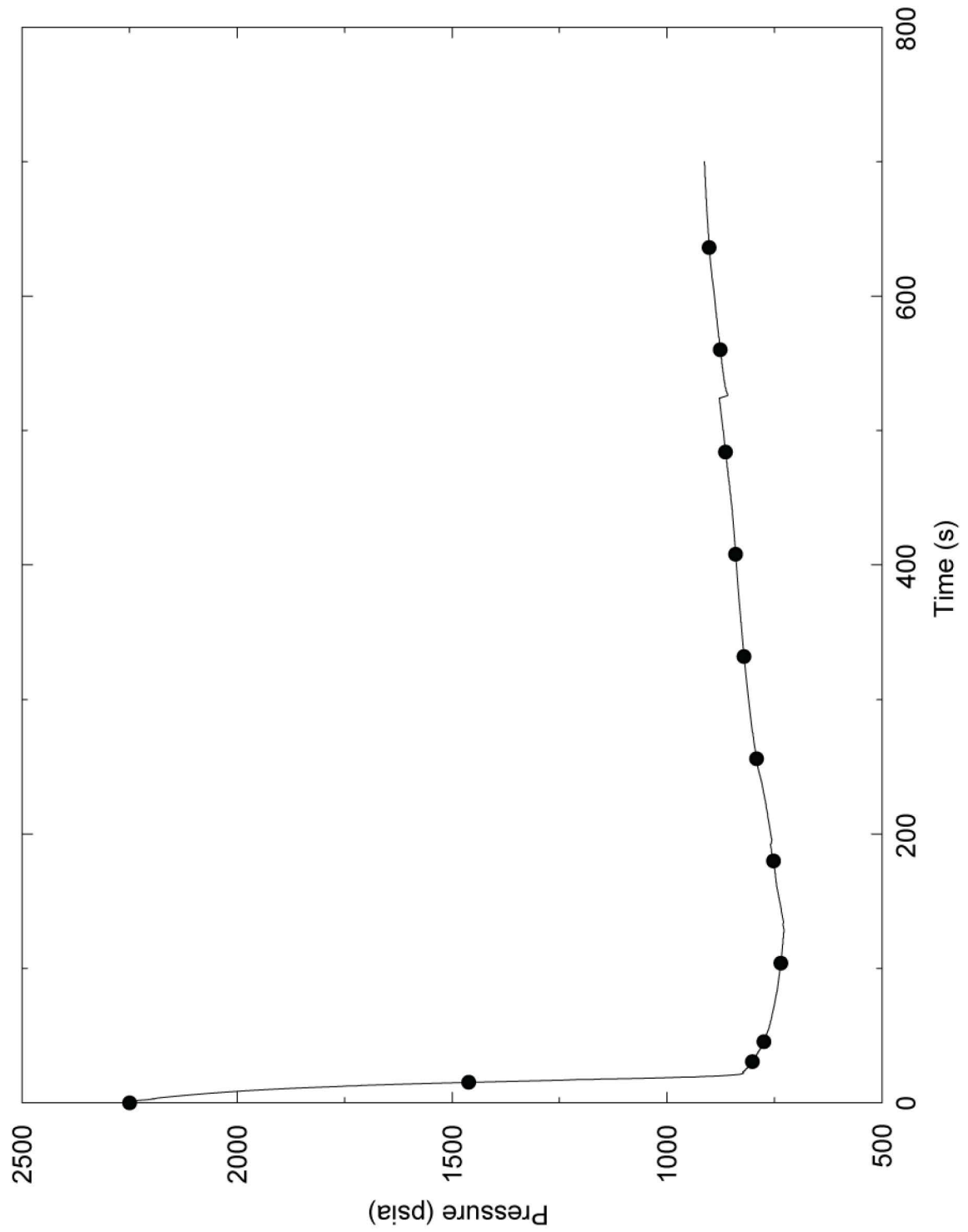
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1**

**PRESSURIZER LIQUID LEVEL
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, OFFSITE POWER AVAILABLE)**

FIGURE 15.4.6-16

Pressurizer Pressure – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)



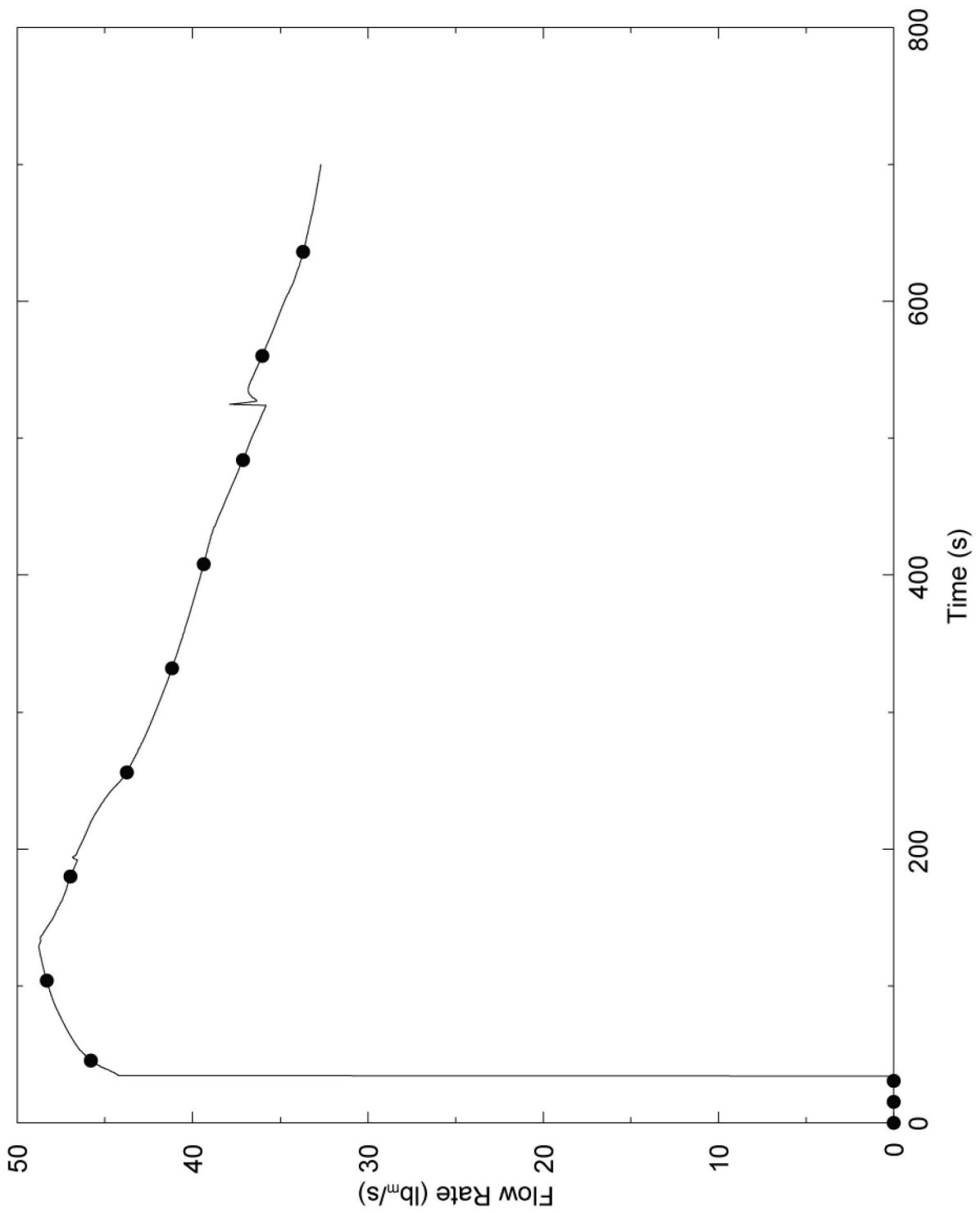
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1**

**PRESSURIZER PRESSURE
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, OFFSITE POWER AVAILABLE)**

FIGURE 15.4.6-17

Total HPSI Flow Rate – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)



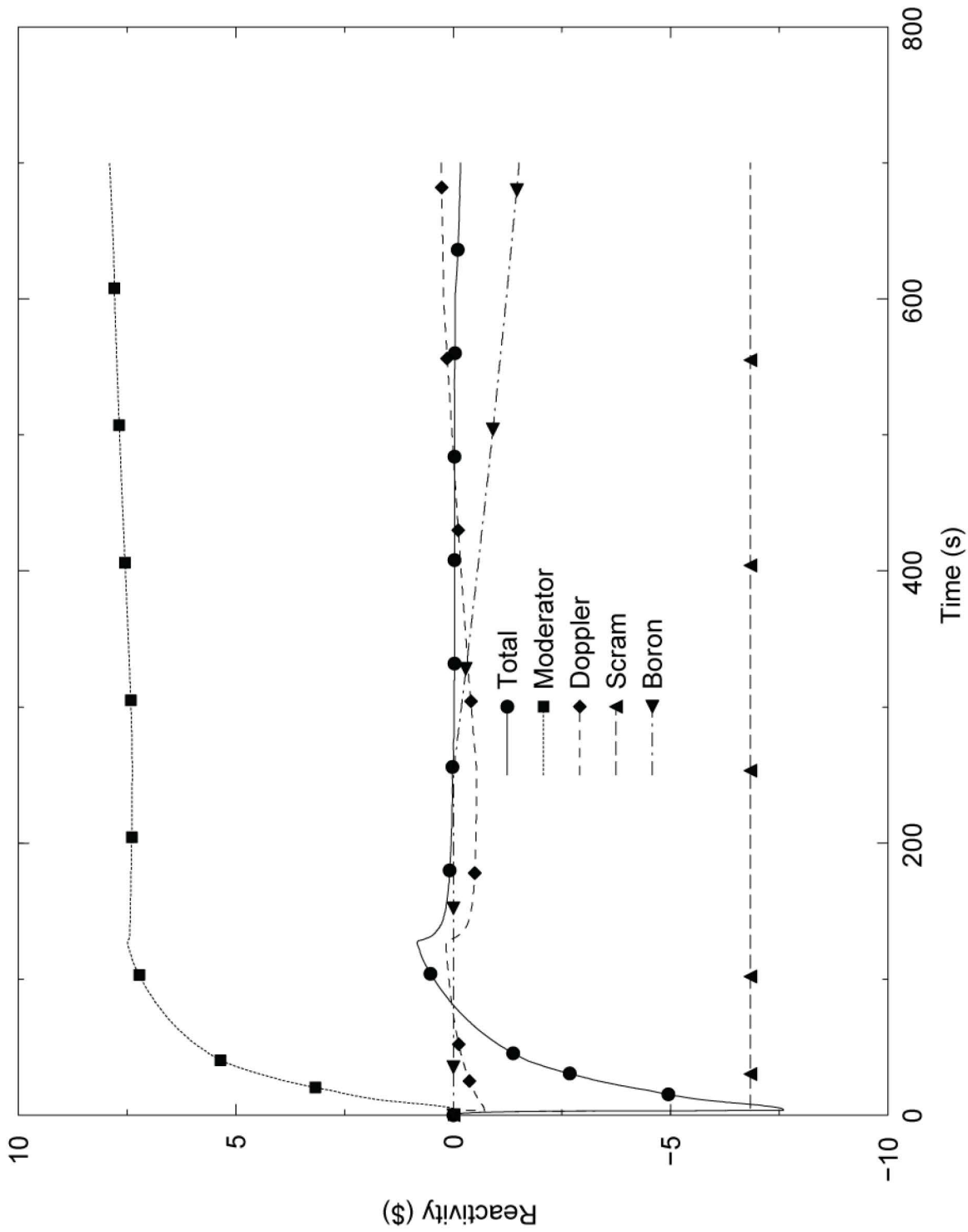
Amendment No. 26 (11/13)

FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1

TOTAL HPSI FLOW RATE
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, OFFSITE POWER AVAILABLE)

FIGURE 15.4.6-18

Reactivity Feedback – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)

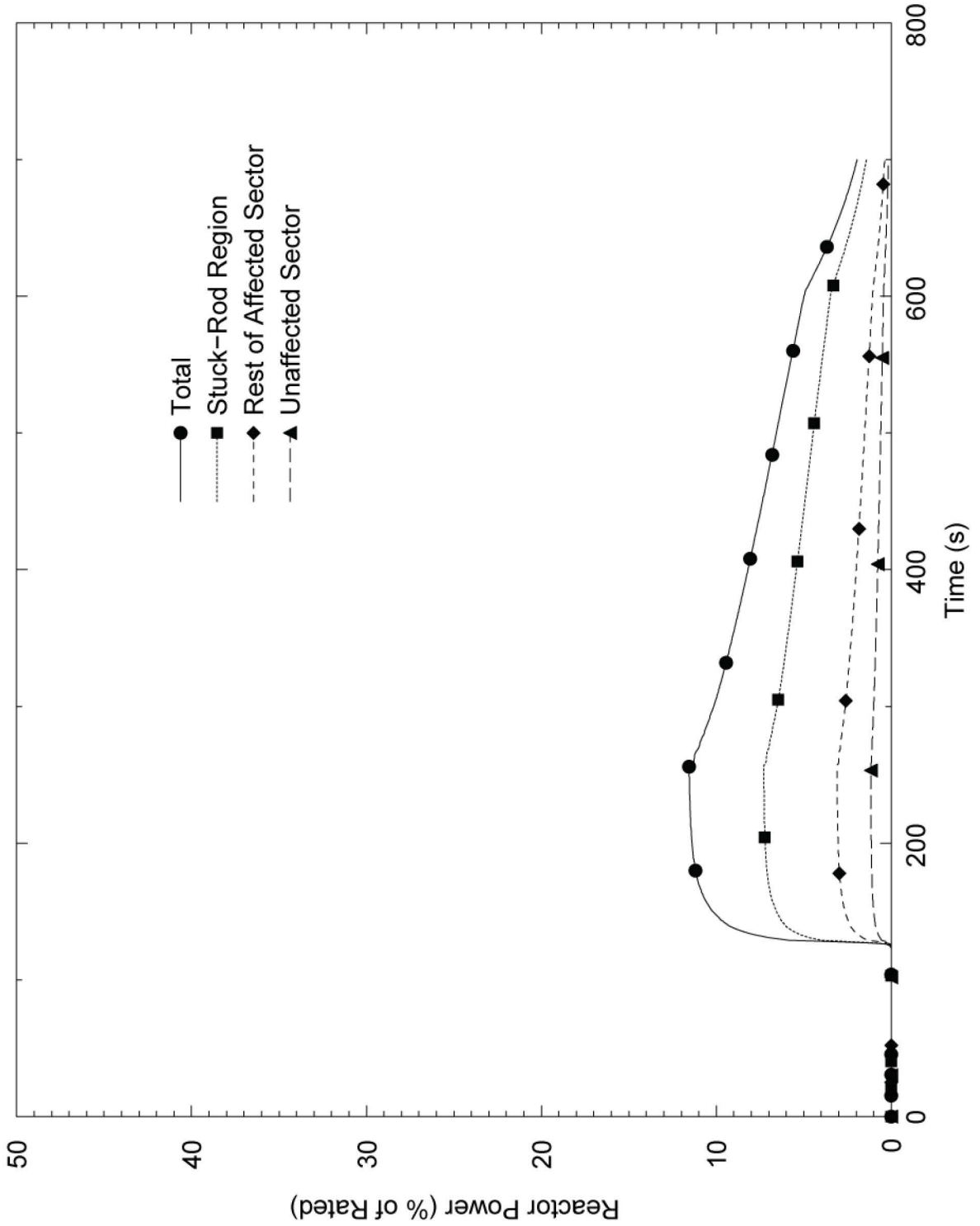


Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1
REACTIVITY FEEDBACK
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, OFFSITE POWER AVAILABLE)**

FIGURE 15.4.6-19

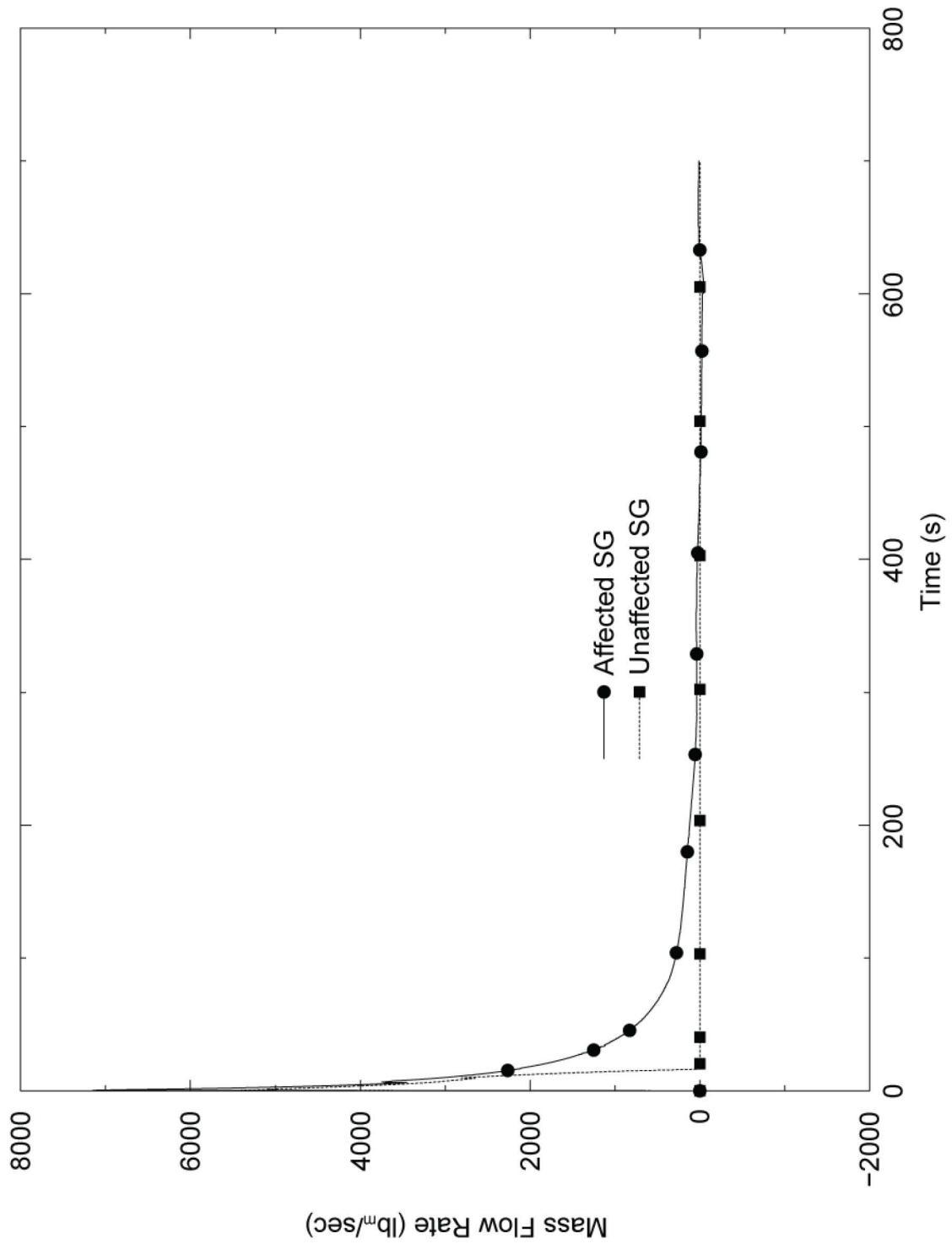
Core Power – Post-Scram Main Steam Line Break (HZP, Offsite Power Available)



Amendment No. 26 (11/13)

<p>FLORIDA POWER AND LIGHT COMPANY ST.LUCIE PLANT UNIT 1</p>
<p>CORE POWER POST-SCRAM MAIN STEAM LINE BREAK (HZP, OFFSITE POWER AVAILABLE)</p>
<p>FIGURE 15.4.6-20</p>

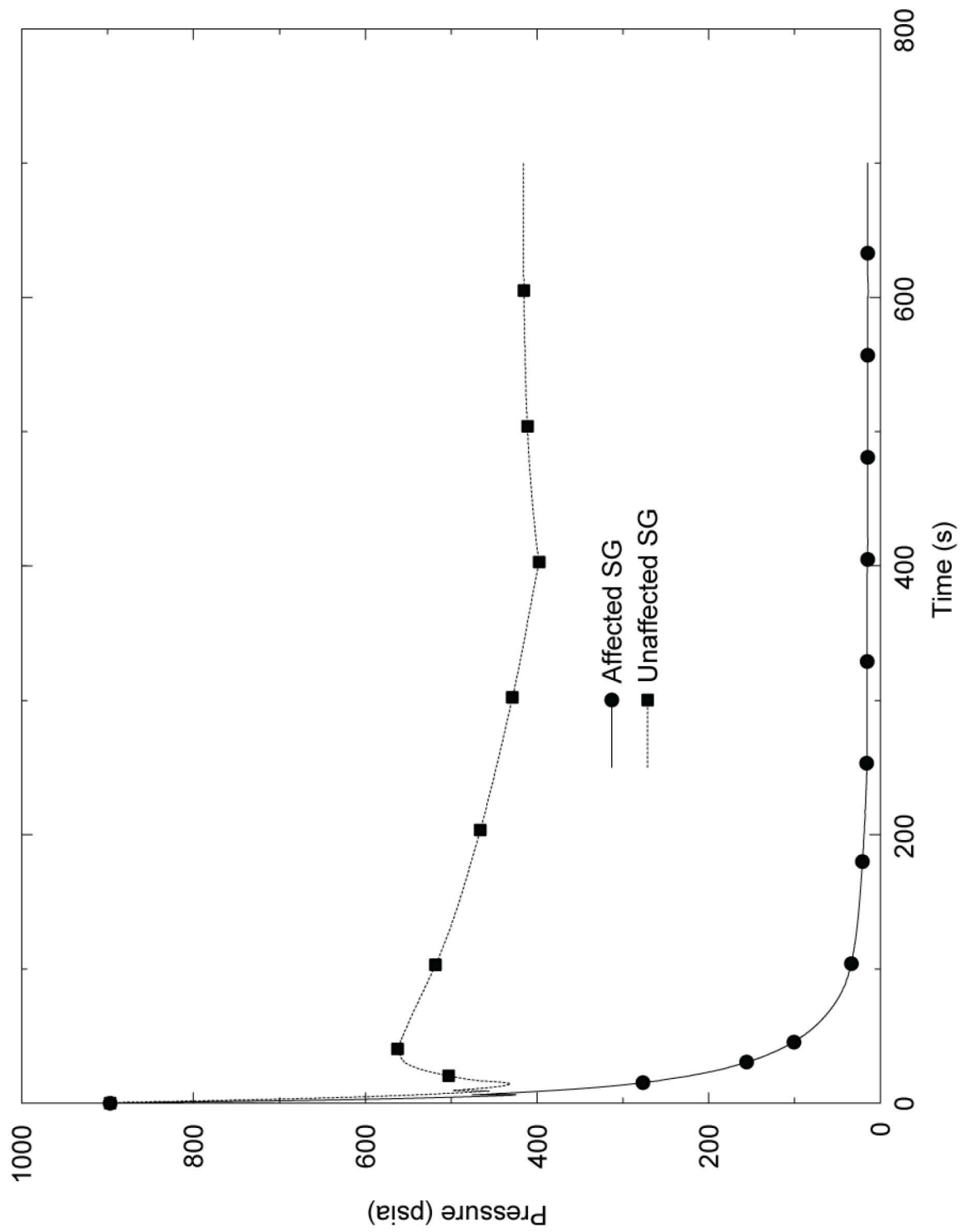
Break Flow Rates – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)



Amendment No. 26 (11/13)

<p>FLORIDA POWER AND LIGHT COMPANY ST. LUCIE PLANT UNIT 1</p>
<p>BREAK FLOW RATES POST-SCRAM MAIN STEAM LINE BREAK (HZP, LOSS OF OFFSITE POWER)</p>
<p>FIGURE 15.4.6-21</p>

Steam Generator Pressures – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)



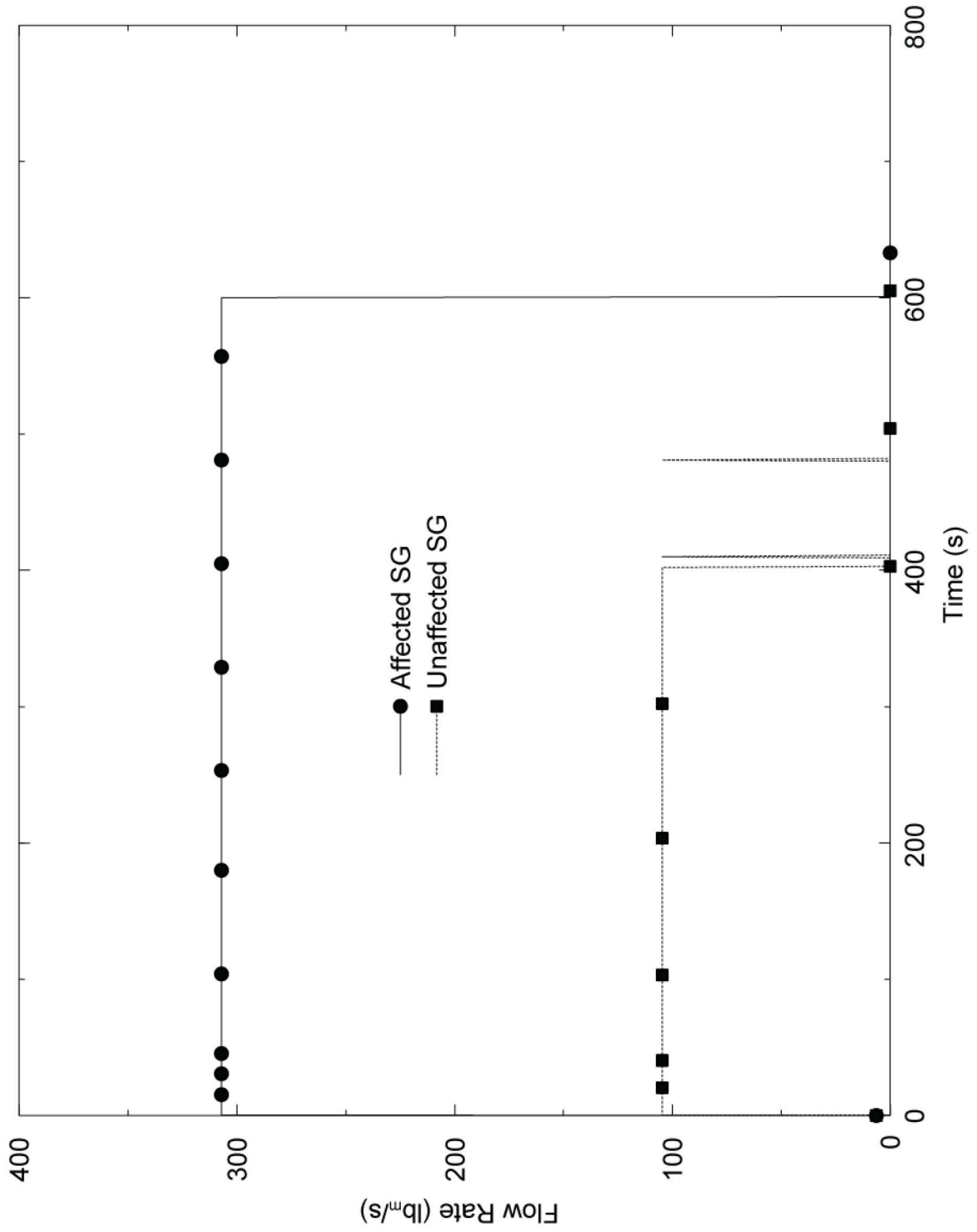
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1**

**STEAM GENERATOR PRESSURES
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, LOSS OF OFFSITE POWER)**

FIGURE 15.4.6-22

Combined MFW and AFW Flow Rates – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)

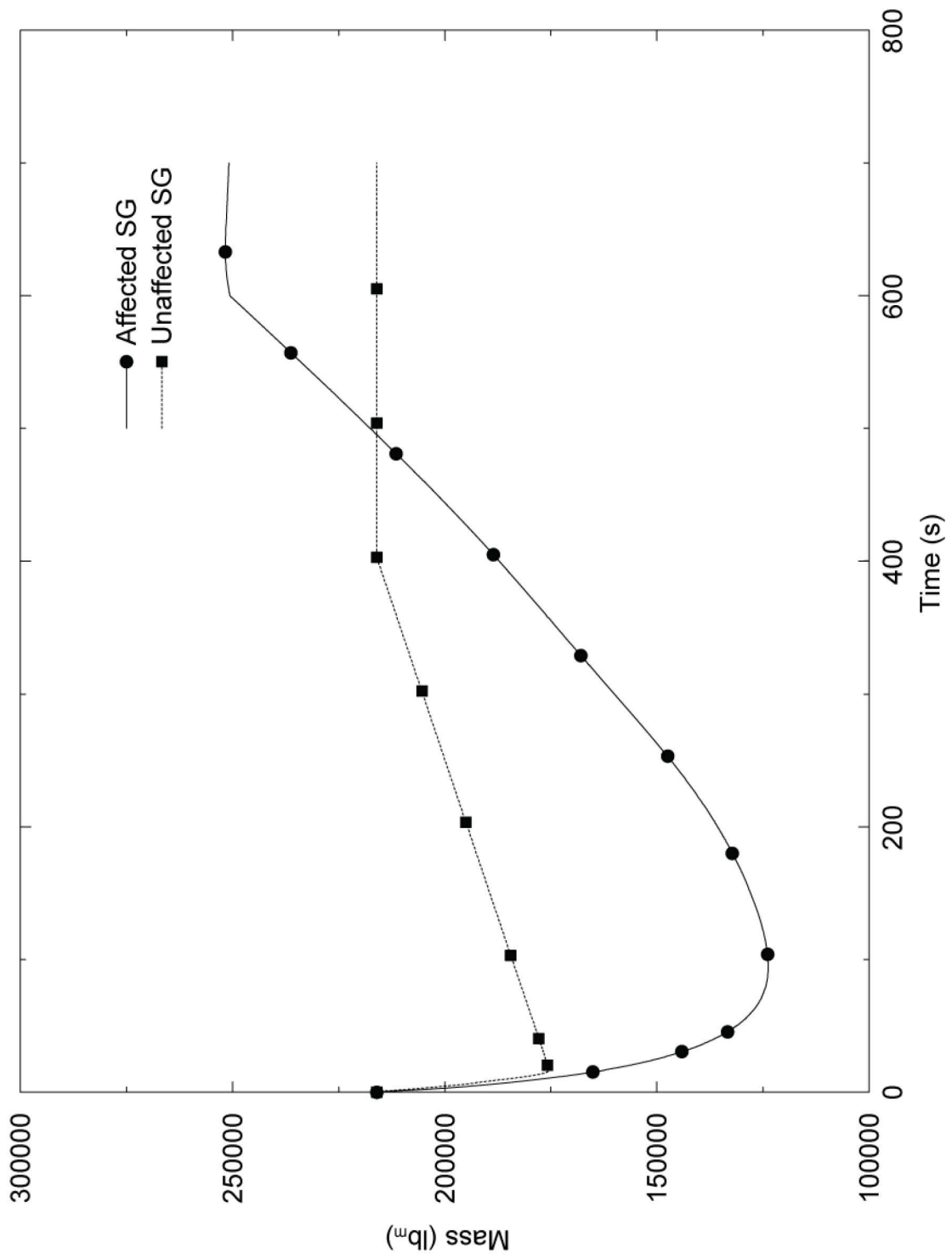


Amendment No. 26 (11/13)

FLORIDA POWER AND LIGHT COMPANY
 ST.LUCIE PLANT UNIT 1
 COMBINED MFW AND AFW FLOW RATES
 POST-SCRAM MAIN STEAM LINE BREAK
 (HZP, LOSS OF OFFSITE POWER)

FIGURE 15.4.6-23

Steam Generator Mass Inventories – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)

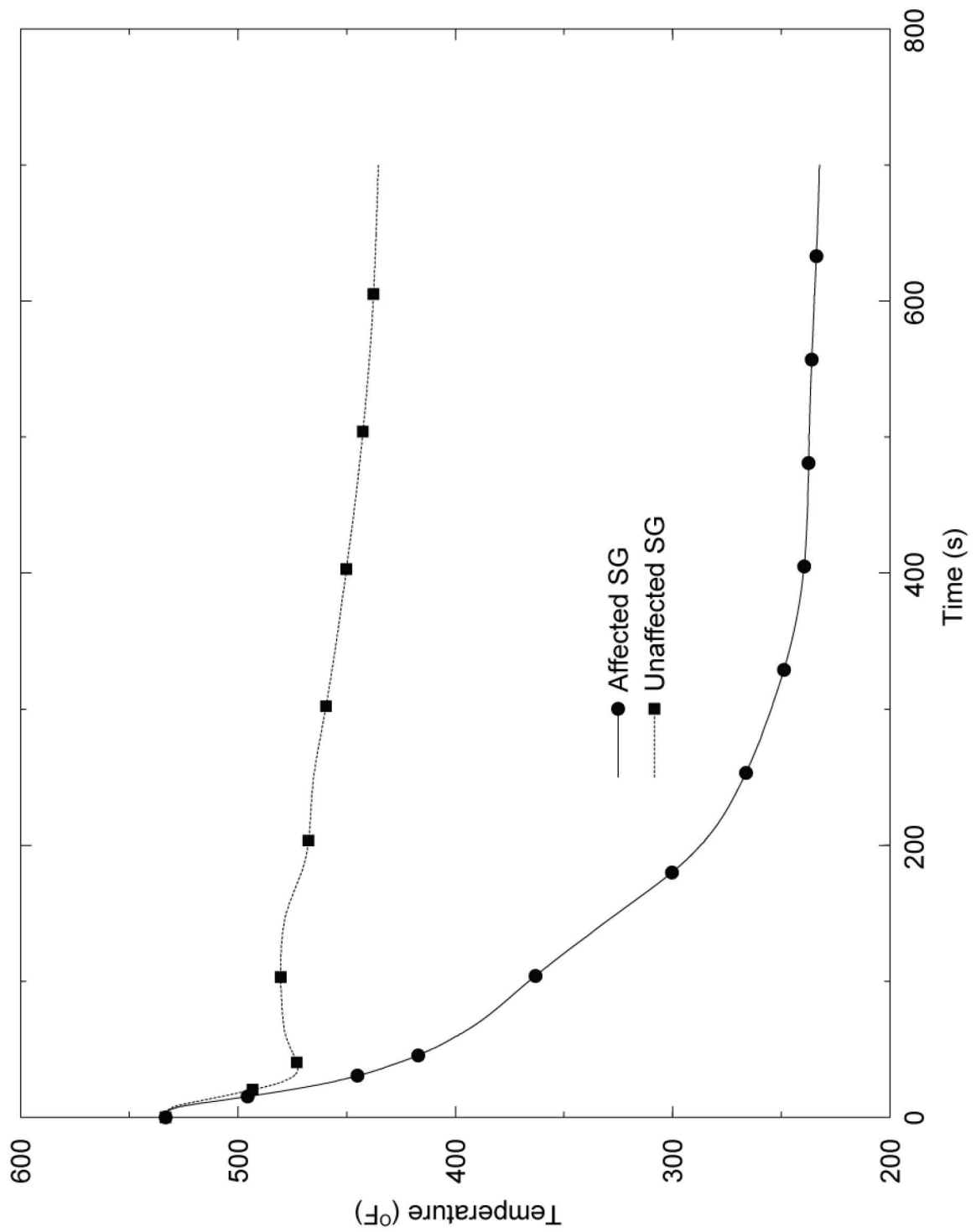


Amendment No. 26 (11/13)

FLORIDA POWER AND LIGHT COMPANY
 ST.LUCIE PLANT UNIT 1
 STEAM GENERATOR MASS INVENTORIES
 POST-SCRAM MAIN STEAM LINE BREAK
 (HZP, LOSS OF OFFSITE POWER)

FIGURE 15.4.6-24

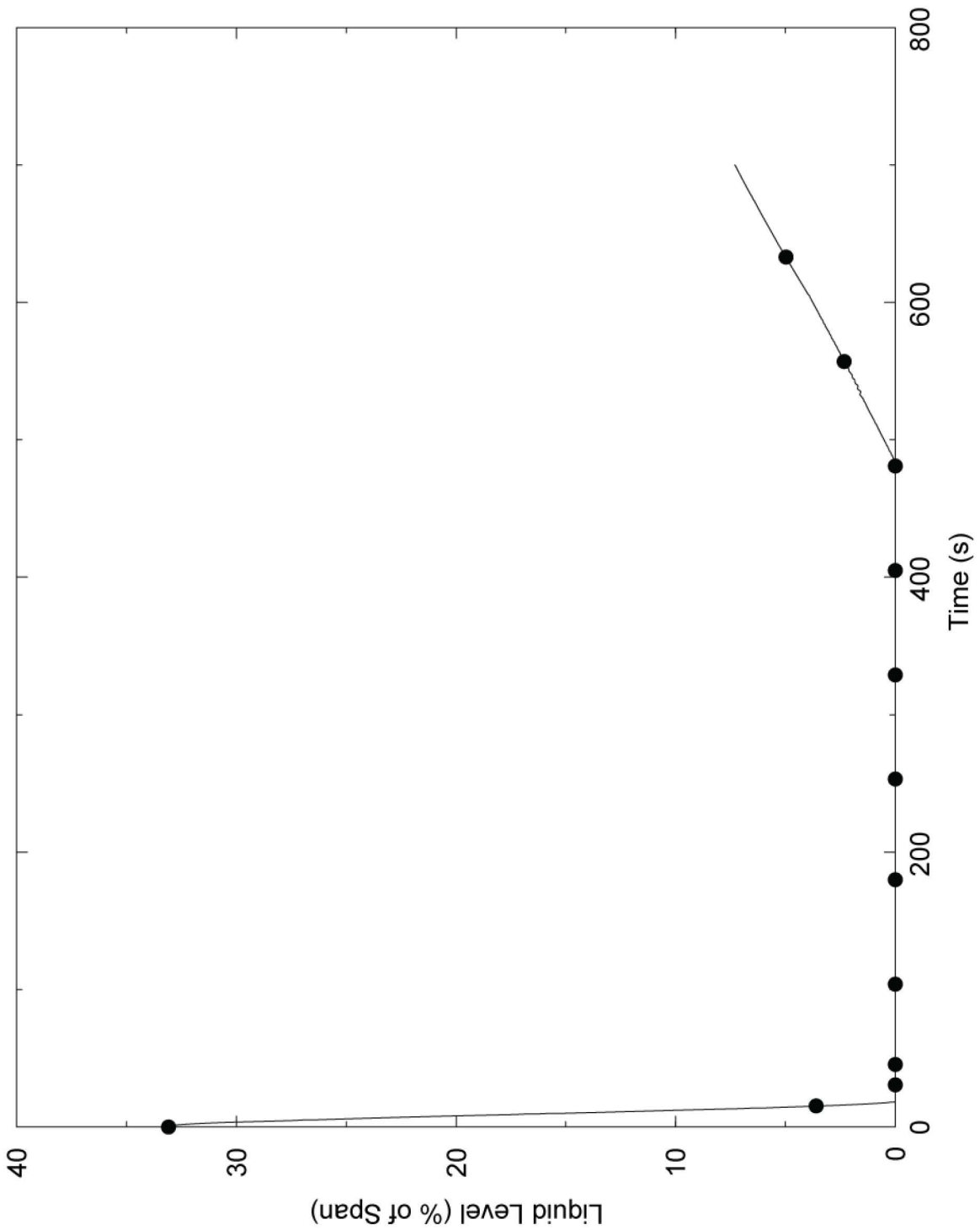
Core Inlet Fluid Temperatures – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)



Amendment No. 26 (11/13)

<p>FLORIDA POWER AND LIGHT COMPANY ST.LUCIE PLANT UNIT 1 CORE INLET FLUID TEMPERATURES POST-SCRAM MAIN STEAM LINE BREAK (HZP, LOSS OF OFFSITE POWER)</p>
<p>FIGURE 15.4.6-25</p>

Pressurizer Liquid Level – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)



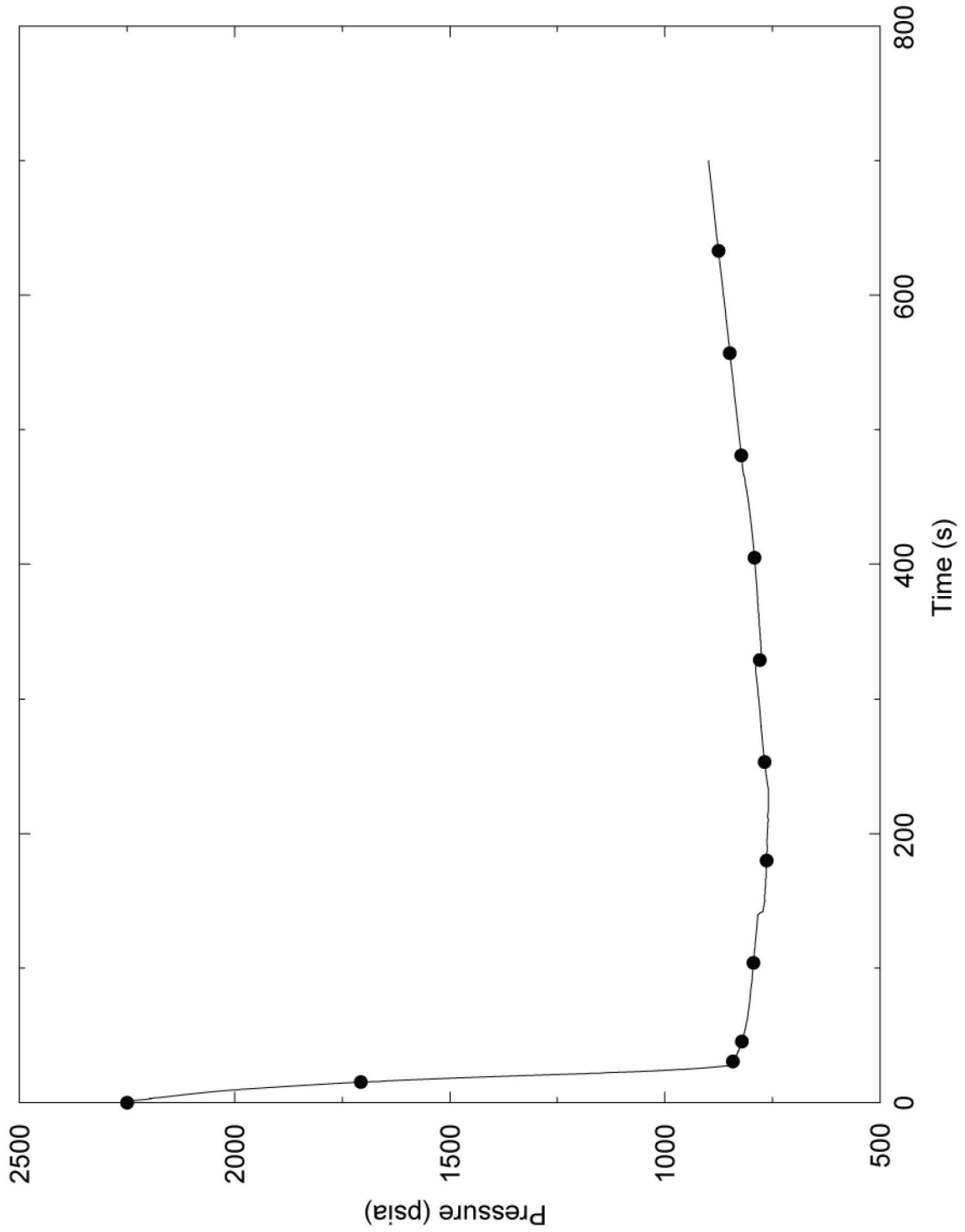
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST.LUCIE PLANT UNIT 1**

**PRESSURIZER LIQUID LEVEL
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, LOSS OF OFFSITE POWER)**

FIGURE 15.4.6-26

Pressurizer Pressure – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)



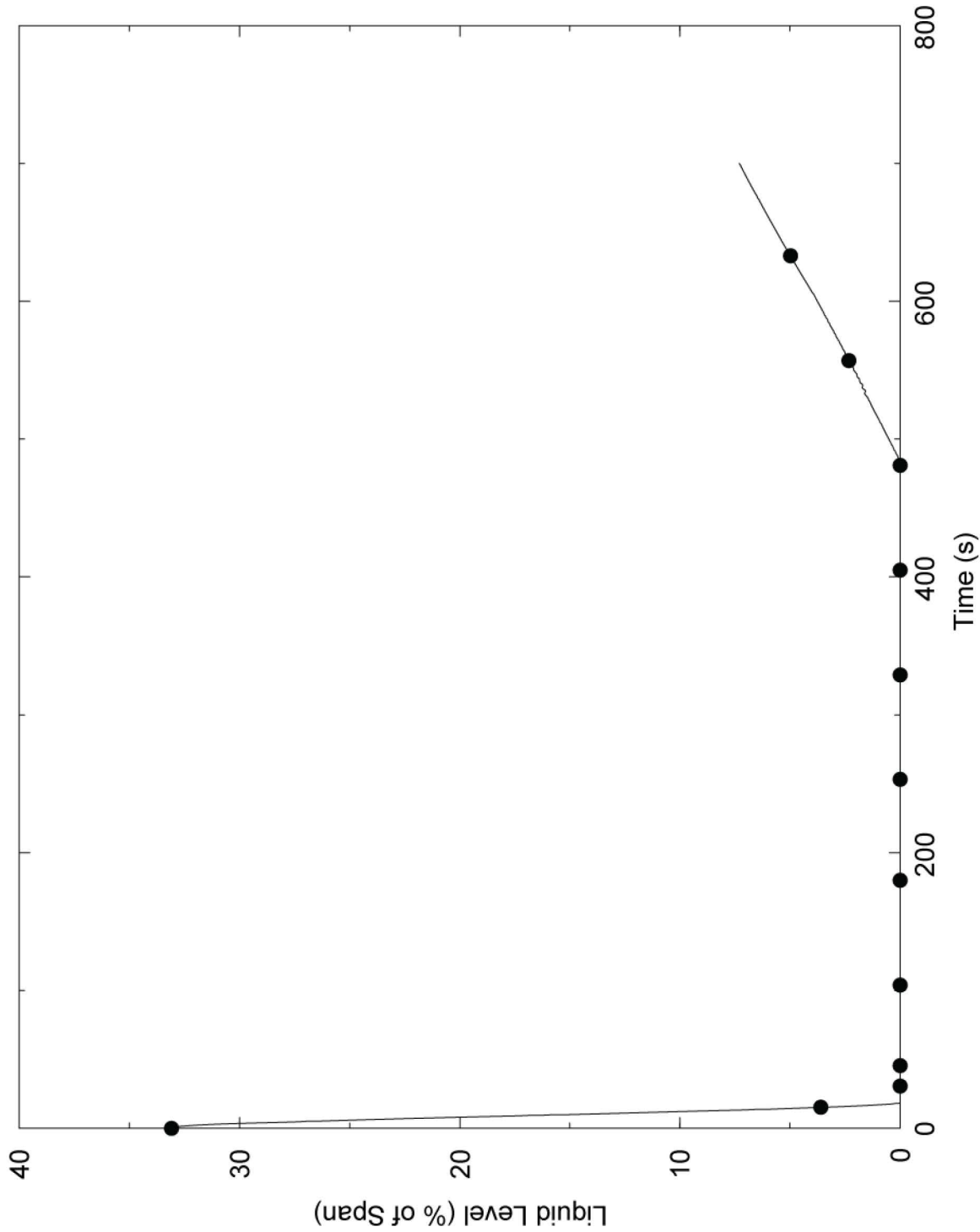
Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**PRESSURIZER PRESSURE
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, LOSS OF OFFSITE POWER)**

FIGURE 15.4.6-27

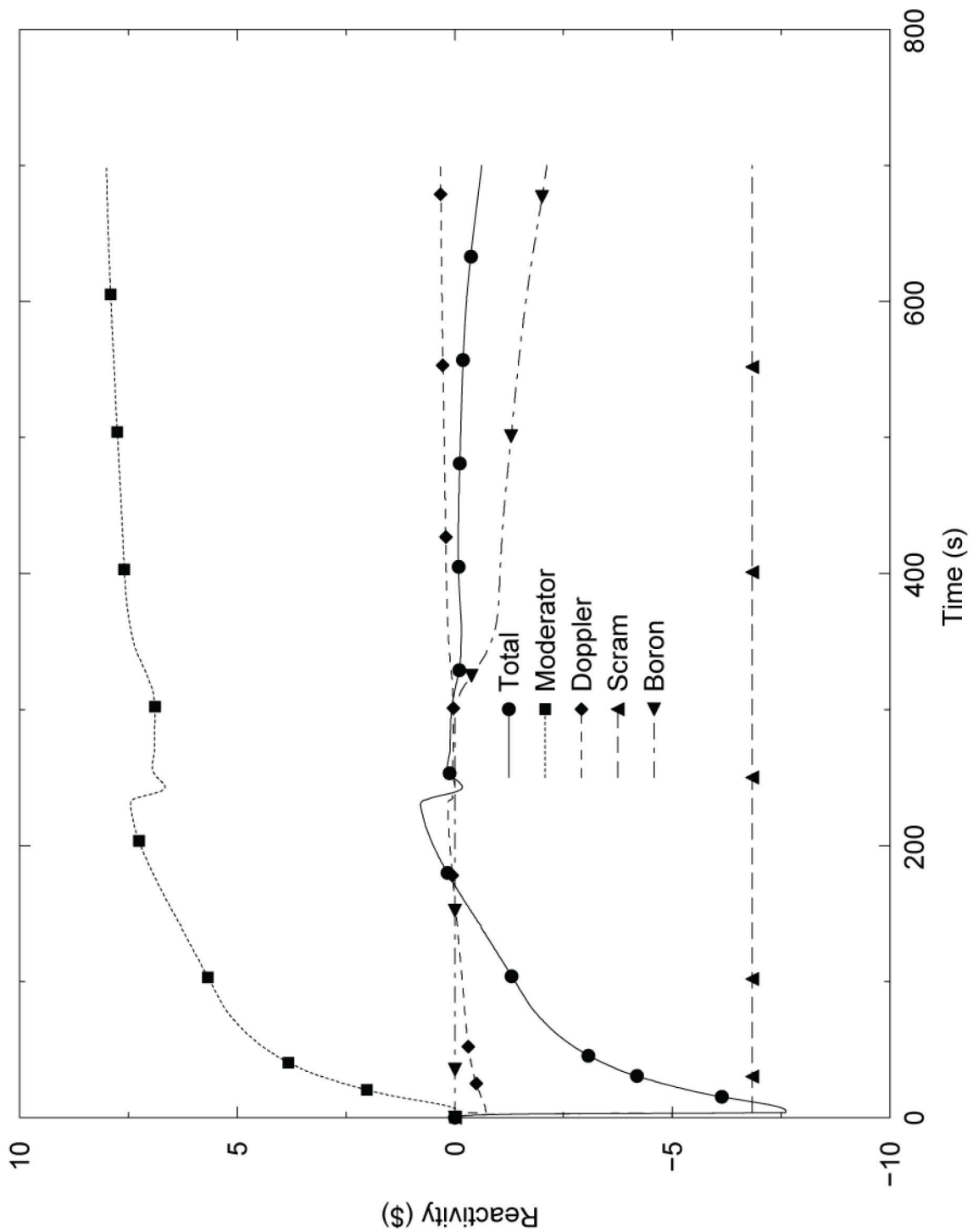
Pressurizer Liquid Level – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)



Amendment No. 26 (11/13)

<p>FLORIDA POWER AND LIGHT COMPANY ST.LUCIE PLANT UNIT 1</p>
<p>TOTAL HPSI FLOW RATE POST-SCRAM MAIN STEAM LINE BREAK (HZP, LOSS OF OFFSITE POWER)</p>
<p>FIGURE 15.4.6-28</p>

Reactivity Feedback – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)

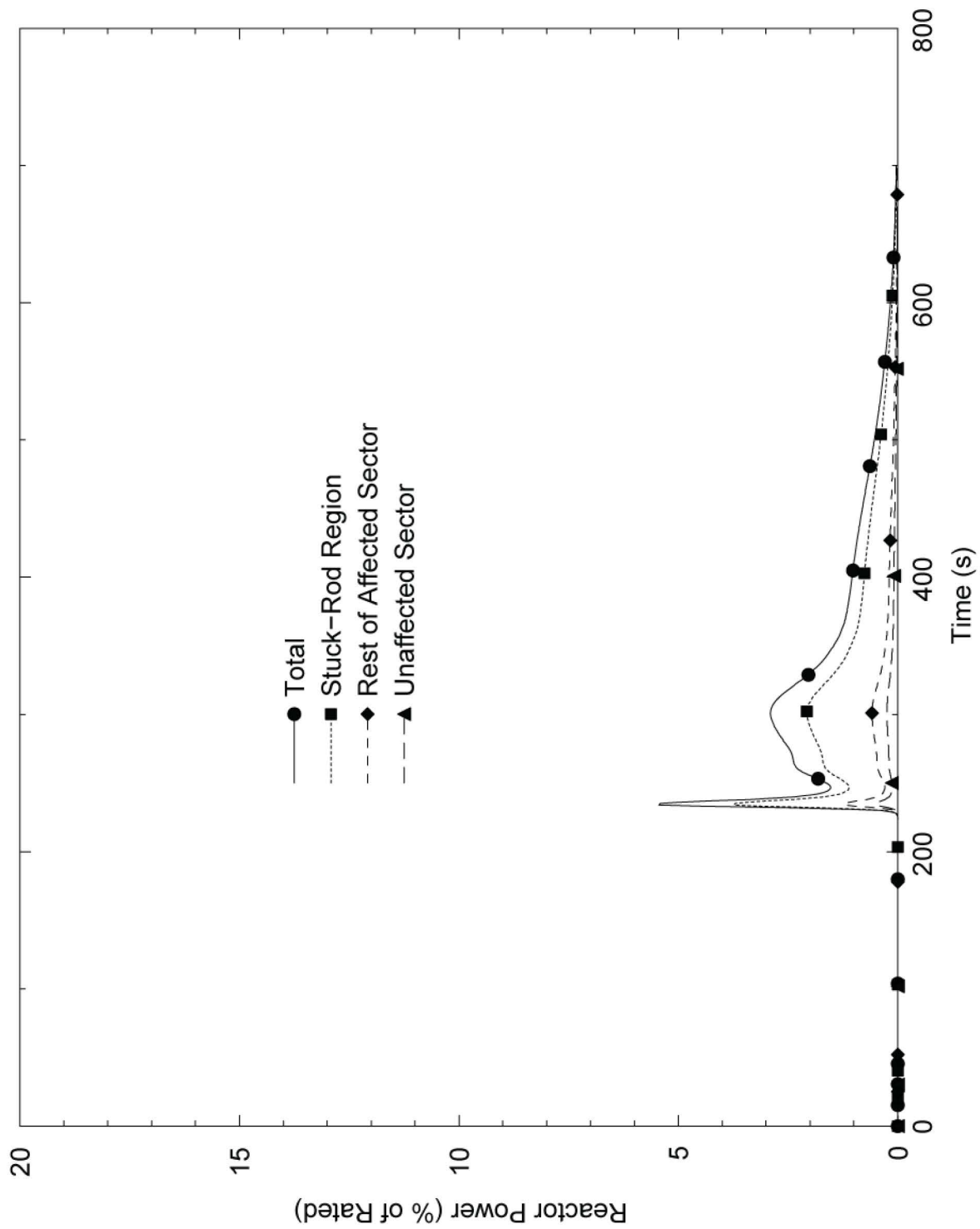


Amendment No. 26 (11/13)

FLORIDA POWER AND LIGHT COMPANY
 ST.LUCIE PLANT UNIT 1
 REACTIVITY FEEDBACK
 POST-SCRAM MAIN STEAM LINE BREAK
 (HZP, LOSS OF OFFSITE POWER)

FIGURE 15.4.6-29

Core Power – Post-Scram Main Steam Line Break (HZP, Loss of Offsite Power)



Amendment No. 26 (11/13)

**FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1**

**CORE POWER
POST-SCRAM MAIN STEAM LINE BREAK
(HZP, LOSS OF OFFSITE POWER)**

FIGURE 15.4.6-30

15.5 REFERENCES FOR CHAPTER 15

1. Florida Power & Light Company, St. Lucie Plant Unit No. 1, Final Safety Analysis Report (FSAR), current through Revision 57.
2. R. E. Uhrig (FP&L) to D.G. Eisenhut (NRC), St. Lucie Unit 1, Docket 50-335 Proposed Amendment to Facility Operating License DPR-67, FP&L letter L-80-381 dated November 14, 1980.
3. R. E. Uhrig (FP&L) to D. G. Eisenhut (NRC), St. Lucie Unit 1, Docket 50-335, Proposed Amendment to Facility Operating License DPR-67, Shutdown Margin, Main Steam Isolation System Changes, and Control Element Assembly Sleeving, FP&L letter L-81-306 dated July 23, 1981.
4. R. E. Uhrig (FP&L) to R. A. Clark (NRC), St. Lucie Unit 1, Docket 50-335 Stretch Power Proposed Amendment, FP&L letter L-81-388 dated September 4, 1981.
5. Styrikovich, M. A., Martynova, I. I., Kathovskaya, K., Ya., Dubrovskii, I. Ya., and Smirnova, I. N., "Transfer of Iodine from Aqueous Solutions to Saturated Vapor," Translated from Atomnaya Energija, Vol. 17.
6. R. V. MacBeth, "An Appraisal of Forced Convection Burn-Out Data," Proc. Instu. Mech. Engrns, 1965-66, Vol 180, Pt. 3C, pp. 37-50.
7. CENPD-199-P, "CE Setpoint Methodology," April 1976.
8. Engineering Evaluation PSL-ENG-SENS-25-001, Rev. 0, ST. LUCIE 1 AND 2 UFSAR SAFETY ANALYSIS REFERENCES.
9. DELETED
10. DELETED
11. DELETED
12. DELETED
13. DELETED
14. CENPD-183, "CE Methods for Loss of Flow Analysis," July 1975.
15. Letter, Robert E. Uhrig, (FP&L) to Victor Stello (NRC), dated February 22, 1979, "St. Lucie Unit 1 docket No. 50-335 Proposal Amendment to Facility Operating License DPR-67."

EC300419

REFERENCES FOR CHAPTER 15 (Cont'd)

16. Letter, A. E. Lundvall (BG&E), to R. Reid (NRC), dated February 23, 1979, "Calvert Cliffs Unit 1 License Amendment," Docket No. 50-317, DPR-53.
17. Letter, W. G. Council (NNECO) to R. Reid (NRC), dated February 12, 1979, "Millstone Point-2 License Amendment, Power Upgrading," Docket No. 50-336, DPR-65.
18. DELETED
19. CEN-124 (B)-P, "Statistical Combination of Uncertainties, Part 2," January 1980.
20. DELETED
21. DELETED
22. DELETED
23. DELETED
24. DELETED
25. DELETED
26. DELETED
27. Notley, M. J. F., et. al., Zircaloy Sheathed UO₂ Fuel Elements Irradiated at Values of K_d Between 40 and 83 W/cm, AECL-1676, December 1962.
28. MacDonald, R. D., and Bain, A. S., Irradiation of Zircaloy-2 Clad UO₂ to Study Sheath Deformation, AECL-1685, December 1962.
29. Bain, A. S., et. al., Trefoil Bundles of NPD 7 - Element Fuel Irradiated to 9100 Mwd/tonne U, AECL-1895, January 1964.
30. MacDonald, R. D., and Parry, G. W., Zircaloy-2 Clad UO₂ Fuel Elements Irradiation to a Burnup of 10,000 Mwd/tonne U, AECL-1952, January 1964.
31. Notely, M. J. F., and MacEwan, J. R., The Effect of UO₂ Density on Fission Product Gas Release and Sheath Expansion, AECL-2230, March 1965.

REFERENCES FOR CHAPTER 15 (Cont'd)

32. Parker, G. W., et. al., Prompt Release of Fission Products From Zircaloy Clad UO₂ Fuel, Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1967, ORNL-4228, pages 3-24.
33. Davis, D., Long, G., and Stanaway, W. P., Emission of Volatile Fission Products from Uranium Dioxide, AERE-4-4342, June 1963.
34. Diffey, H. R. , et al. , "Iodine Cleanup in a Steam Suppression System," International Symposium on Fission Product Release and Transport Under Accident Conditions, Oak Ridge, Tennessee, CONF-650407, Vol. 2, pages 776-804 (1965).
35. Dadillon, Jr., "Efficacite des Pieges A Charbon Impregne vis-a-vis des Produits de Fission Degages Lors D'un Accident de Pile Piscine," Paper SM-110/10, in Proceedings Treatment of Airborne Radioactive Wastes Symposium, New York, 1968, IAEA.
36. DELETED
37. Shapiro, Dr. Z. M. and Moffette, T. R., "Hydrogen Flammability Data and Application to PWR Loss of Coolant Accident," WAPD-Sc-545, September 1957.
38. Cottrell, Wm. B. and Savolainen, A. W., (Editors), "U.S. Reactor Containment Technology," Vol. I (Chapter 5), ORNL-NSIC-5, August 1965.
39. Coward, H. F., and Jones, G. W., "Limits of Flammability of Gases and Vapors," Bureau of Mines Bulletin 503, 1952.
40. Not used.
41. USAEC Report DP-778, "Application of Activated Carbon in Reactor Containments," G. H. Prigge, September 1962.
42. R. E. Uhrig (FPL) to D. K. Davis (NRC) Re: St. Lucie Unit No. 1, Docket No. 50-335, Boron Dilution, L-77-370 dated December 8, 1977.

REFERENCES FOR CHAPTER 15 (Cont'd)

43. DELETED
44. DELETED
45. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," XN-74-5, Rev. 1, Exxon Nuclear Company, Inc., Richland, Washington 99352, May 1975.
46. Letter, T. P. Gates to C. G. O'Farrill (FPL), "Transmittal of St. Lucie Unit 1 Flow Coastdown Data," F-CE-7775, October 25, 1982.
47. St. Lucie Unit 1 Technical Specifications, Appendix A to License #DPR-67, as per Amendment 46, November 3, 1981.
48. Moody, F. J., ASME Transactions, p. 134, February 1965.
49. Florida Power & Light Company, St. Lucie Plant, Unit No. 1, Updated Final Safety Analysis Report.
50. DELETED
51. DELETED
52. DELETED
53. DELETED
54. XN-NF-82-98, "St. Lucie Unit 1 LOCA Analysis Using the EXEM/PWR ECCS Model," Exxon Nuclear Company, Inc., January 1983.
55. XN-NF-83-83 (P), "Re-Evaluation of Plant Transients for St. Lucie Unit 1 With Decreased Core Flow."
56. DELETED
57. DELETED
58. "Design and Safety Report for St. Lucie Unit 1 Cycle 4 at 2700 MWth," Attachment 3 to Operating License DPR-67, November 14, 1980.
59. Moody, F. J., ASME Transactions, p. 134, February 1965.

REFERENCES FOR CHAPTER 15 (Cont'd)

60. XN-NF-86-23, Revision 1, "St. Lucie Unit 1 LOCA/ECCS Analysis with 11% Steam Generator Tube Plugging," Exxon Nuclear Company, March 1986.
61. DELETED
62. XN-NF-84-85(P), Revision 1, "St. Lucie Unit 1 Radiological Assessment of Postulated Accidents," Exxon Nuclear Company, April 1985.
63. Safety Evaluations by the Office of Nuclear Reactor Regulation Related to Amendment No. 63 to Facility Operating License No. DPR-67, Florida Power and Light Company, St. Lucie Unit 1, Docket No. 50-335.
64. XN-NF-84-93(P), "Steamline Break Methodology for PWRs," Exxon Nuclear Company, Richland, WA, November 1984.
65. XN-NF-85-73, Revision 2, "St. Lucie Unit 1 Cycle 7 Safety Analysis Report," Exxon Nuclear Company, Richland, WA, October 1985.
66. Energia Nucleare, Volume 14, No. 9, September 1967, "Studies on Burnout, Part 3 - A New Correlation for Round Ducts and Uniform Heating and its Comparison with World Data, L. Biasi et al." EC292761
67. Carolina Power and Light Company, H.B. Robinson Steam Electric Plant, Unit No. 2 Amendment to Facility Operating License, Docket No. 50-261, Amendment No. 71, License No. DPR-23, USNRC, Washington D.C.
68. XN-NF-86-140(P), "Review and Analysis of Chapter 15 Events for St. Lucie Unit 1 with 15% Steam Generator Tube Plugging," March 1987.
69. "St. Lucie Unit 1 Updated Final Safety Analysis Report, Amendment 4," Florida Power and Light Company, July 1982.
70. XN-NF-86-137, "St. Lucie Unit 1 LOCA/ECCS Analysis with 15% Steam Generator Tube Plugging," November, 1986.
71. ANF-87-148, "St. Lucie Unit 1 LOCA/ECCS Extended Exposure Analysis," November 1987.
72. Letter, T J Helbling (ANF) to A R Morse (FPL) TJH:157:88, dated June 13, 1988.
73. XN-NF-86-169, Revision 1, "St. Lucie Unit 1 Cycle 8 Safety Analysis Report," Exxon Nuclear Company, February 1987.
74. ANF-88-029, "St. Lucie Unit 1 Cycle 9 Safety Analysis Report," Advanced Nuclear Fuels Corporation, April 1988.
75. DELETED
76. Letter, J A Norris (NRC) to J H Goldberg, "St. Lucie Unit 1 - Issuance of Amendment Re: Automatic Feedwater Actuation System Setpoints (TAC No. 76172)," dated September 11, 1990.

REFERENCES FOR CHAPTER 15 (Cont'd)

77. DELETED
78. ANF-89-120, "St. Lucie Unit 1 Cycle 10 Safety Analysis Report," January 1990.
79. Letter, G.M. Holahan (NRC) to R. A. Copeland (SPC), "Acceptance for Referencing of the Topical Report XN-NF-82-49 (P) Revision 1, Supplement 1, "Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break LOCA Model CTAC No. M83302)," October 3, 1994.
80. DELETED
81. DELETED
82. DELETED
83. DELETED
84. DELETED
85. DELETED
86. Letter, T.J. Helbling to J.D. Mantyh, TJH:019:90, "Response to Comments on St Lucie Unit 1, Cycle 10 Safety Analysis," dated January 30, 1990.
87. St. Lucie Unit 1 Large Break LOCA/ECCS Analysis, with 25% +/- 7% SGTP, EMF-92-176, Siemens Nuclear Power Corporation, Richland, WA 99352, February 1993.
88. Letter, J.L. Holm (SNP) to W.L. Parks (FPL), "Transmittal of LOEL Plots for Incorporation into St. Lucie Unit 1 UFSAR," JLH:048:91, February 22, 1991.
89. "St. Lucie Unit 1 Cycle 11 Safety Analysis Report," SNP 91-150, Siemens Nuclear Power Corporation, Richland, WA 99352, September 1991.
90. Letter from J.A. Norris to J. H. Goldberg, "St. Lucie, Units 1 and 2 - 10 CFR 50.63 - Station Blackout (TAC Nos. 68608 and 68609)," September 12, 1991, Docket Nos. 50-335 and 50-389.
91. DELETED
92. DELETED

EC292761

REFERENCES FOR CHAPTER 15 (Cont'd)

93. DELETED
94. FPL Letter L-89-326 to the NRC, "Turkey Point Units 3 & 4 Dockets Nos. 50-250 and 50-251 - St. Lucie Units 1 and 2 Docket Nos. 50-335 and 50-389, Report NTH-TR-01, RETRAN MODEL QUALIFICATION," October 2, 1989.
95. NUMARC 87-00, Rev. 1, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors.
96. Letter from J. A. Norris to J. H. Goldberg, "St. Lucie Plant, Units 1 and 2 - Response to 10 CFR 50.63 - Station Blackout," dated June 11, 1992.
97. "St. Lucie Unit 1 Cycle 14 Safety Analysis Report, (FSAR Chapter 15 Review and Mechanical)," EMF-96-026, Siemens Power Corporation - Nuclear Division, April 1996.
98. "St. Lucie Unit 1 Chapter 15 Event Review and Analysis for 25% Steam Generator Tube Plugging," EMF-92-165, Siemens Power Corporation, Nuclear Division, Richland, WA 99352, February 1993.
99. DELETED
100. "St. Lucie Unit 1 Small Break LOCA Analysis," EMF-92-148, Siemens Power Corporation, Nuclear Division, Richland, WA 99352, February 1993.
101. "St. Lucie Unit 1 Loss-of-Feedwater Transient With Degraded Auxiliary Feedwater Flow," EMF-2515, Revision 0, Siemens Power Corporation, Nuclear Division, December 2000.
102. DELETED
103. "Reactor Analysis System for PWRs, Volume 1 - Methodology Description, Volume 2 - Benchmarking Results," EMF-96-029 (P)(A), Volumes 1 and 2, Siemens Power Corporation, January 1997.
104. USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Plants," July 2000.
105. NUREG/CR-5950, "Iodine Evolution and pH Control."
106. NRC Regulatory Issue Summary 2006-04, "Experience With Implementation Of Alternative Source Terms," March 7, 2008.
107. NAI-1101-043, "AST Licensing Technical Report for St. Lucie Unit 1" (part of NRC submittal L-2007-085, "St. Lucie Unit 1 Proposed License Amendment Alternative Source Term and Conforming Amendment").
108. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983.
109. EMF-1961(P)(A) Revision 0, *Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors*, Siemens Power Corporation, July 2000.

REFERENCES FOR CHAPTER 15 (Cont'd)

110. EMF-92-153(P)(A) Revision 1, *HTP: Departure From Nucleate Boiling Correlation for High Thermal Performance Fuel*, Siemens Power Corporation, January 2005.
111. XN-NF-78-44(NP)(A), *A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors*, Exxon Nuclear Company, October 1983.
112. ANP-2902(P), Revision 2, *St. Lucie Unit 1 EPU Non-LOCA Safety Analysis Engineering Report for Zr-4 Fuel Design*, April 2011.
113. EMF-2328(P)(A) Revision 0, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*, Framatome ANP, Inc., March 2001 (as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU - Information to Support License Amendment Request," Revision 0).
114. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal- Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.
115. XN-NF-81-58(P)(A) Revision 2 Supplements 3 and 4, *RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model*, Advanced Nuclear Fuels, June 1990.
116. XN-75-21(P)(A) Revision 2, *XCOBRA-1/IC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation*, Exxon Nuclear Company, January 1986.
117. EMF-2310(P)(A) Revision 1, SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, Framatome ANP, Inc., May 2004 (as supplemented by ANP-3000(P), "St. Lucie Unit 1 EPU -Information to Support License Amendment Request," Revision 0).
118. BAW-10240 (P)(A), Revision 0, *Incorporation of M5TM Properties in Framatome ANP Methods*, Framatome ANP, Inc., May 2004.
119. DBD-FP-1 , Fire Protection Design Basis Document.
120. EMF-2310(P)(A) Revision 1, Supplement 2, Revision 0, SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors, Framatome, Inc. March 2023.
121. ANP-4073 Revision 0, St. Lucie Unit 1 Cycle 32 Safety Analysis Report, Framatome, Inc. December 2023.

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15.6 SUMMARY OF OPERATING LIMITS

Operating limits for the St. Lucie Unit 1 nuclear plant are summarized below. Methods of analysis for determining or verifying the operating limits are detailed in Subsection 15.6.5 and Reference 1.

15.6.1 REACTOR PROTECTION SYSTEM

The reactor protection system (RPS) is designed to assure that the reactor is operated in a safe and conservative manner. The input parameters for the RPS are denoted as limiting safety system settings (LSSS). The values or functional representation of the LSSSs are calculated to ensure adherence to the specified acceptable fuel design limits (SAFDLS) during steady state and anticipated operational occurrences (AOOs). The safe operation of the reactor is also maintained by restricting reactor operation to be in conformance with the limiting conditions for operation (LCOS) which are administratively applied at the reactor plant. The LSSS and LCO parametric values are presented in the following sections.

15.6.2 SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS

The SAFDLs are experimentally or analytically based limits on the fuel and cladding which preclude fuel damage. These limits may not be exceeded during steady-state operation or during AOOs. The SAFDLs are used to establish the reactor setpoints to ensure safe operation of the reactor. The specific SAFDLs used to establish the setpoints are:

- The local power density (LPD) which coincides with fuel centerline melt.
- The MDNBR corresponding to the accepted criterion which protects against the occurrence of DNB.

The setpoint verification analysis for the current cycle is performed with a LPD limit corresponding to the maximum LHR that can occur in a fuel rod without the occurrence of fuel centerline melt. It is noted that reload fuel contains gadolinia-bearing fuel rods which, for a given LPD, will operate with a higher fuel temperature and will consequently have a lower LPD limit. These rods are modeled in the centerline melt calculations to ensure that, with a standard fuel rod at the maximum LPD limit, the maximum LPD of the gadolinia-bearing rods will remain far enough below the UO₂ melt limit to prevent centerline melt.

Due to the increased thermal performance of HTP assemblies, as seen in Reference 3, and the fact that the maximum power of the limiting bi-metallic assembly is much less than the peak assembly power, a HTP assembly will be limiting from the standpoint of DNBR.

The HTP critical heat flux correlation was used in the thermal margin analysis with statistical parameters corresponding to an upper 95/95 DNBR limit with an allowance for mixed core penalty. Observance of the limiting conditions for operation will protect against DNB with 95% probability at a 95% confidence level during an AOO.

15.6.3 LIMITING SAFETY SYSTEM SETTINGS

15.6.3.1 Local Power Distribution Control

The local power distribution (LPD) trip limit is the locus of the limiting values of core power level versus axial shape index that will produce a reactor trip to prevent exceeding the fuel centerline melt limit. The correlation between allowed core power level and peripheral axial shape index (ASI) was determined using methods which take into account the total calculated nuclear peaking and the measurement and calculational uncertainties associated with power peaking.

15.6.3.2 Thermal Margin/Low Pressure

The thermal margin/low pressure (TM/LP) trip protects against the occurrence of DNB during steady state operations and for many, but not all, AOOs. This reactor trip system monitors the primary system pressure, core inlet temperature, core power and ASI and a reactor trip occurs when primary system pressure falls below the computed limiting core pressure, P_{var} . As with the LPD trip, a statistical setpoint methodology (Reference 1) is used to verify the adequacy of the existing TM/LP trip. The methodology for the TM/LP trip accounts for uncertainties in core operating conditions, HTP DNB correlation uncertainties, and uncertainties in power peaking. The existing TM/LP trip function for EPU operation at 3020 MWt is given by:

$$P_{var} = 2061 \times A1(ASI) \times QR1(Q) + 15.85 \times T_{in} - 8950,$$

where Q is the higher of the thermal power and the nuclear flux power, T_{in} is the inlet temperature in $^{\circ}F$ and $A1$ and $QR1$ are shown in Figures 15.6-2 and 15.6-3, respectively.

15.6.3.3 Additional Trip Functions

In addition to the LPD and TM/LP trip functions, other reactor system trips have been determined to provide input to the setpoint verification. The setpoints and uncertainties for these trips are shown in Table 15.6-4.

15.6.4 LIMITING CONDITIONS FOR OPERATION

15.6.4.1 DNB Monitoring

The validity of the existing LCO for allowable core power as a function of ASI was verified to ensure adherence to the SAFDL on DNB during a postulated CEA drop and loss-of-flow operational occurrences. The statistical analysis accounted for the effects of uncertainties associated in core operating parameters, the HTP critical heat flux correlation, and power peaking. The allowed core power as a function of ASI for the LCO is provided for the current cycle in the Technical Specifications/COLR.

15.6.4.2 Linear Heat Rate Monitoring

In the event that the incore detector system is not in operation for an extended period of time, the linear heat rate will be monitored through the use of an LPD LCO. The verification of this LCO was performed in a fashion similar to that used in verifying the LPD limiting safety system setting (Section 15.6.3.1). The LPD LCO limits core power so that the linear heat rate LCO based on loss of coolant accident (LOCA) considerations is not exceeded. The LHR heat rate LCO protected by the LPD LCO is depicted in Figure 15.6-7.

15.6.5 SETPOINT ANALYSIS

15.6.5.1 Limiting Safety System Settings

Local Power Distribution

The LPD trip monitors core power and ASI in order to initiate a reactor scram which precludes exceeding fuel centerline melt conditions. In the analysis for this trip function a large number of axial power distributions were examined to establish bounding values of total power peaking, F_Q versus ASI. Axial power distributions and other core neutronics related parameters used in the setpoint verification analyses were generated. Statistical methods were then employed to account for the uncertainties in the parameters that are given in Table 15.6-1.

The allowed power for each ASI was calculated statistically incorporating the uncertainties listed in Table 15.6-1 as described in Reference 1. The results of this calculation verify the adequacy of the LPD LSSS trip function shown in the Technical Specifications/COLR for St. Lucie Unit 1.

Thermal Margin/Low Pressure LSSS

The thermal margin/low pressure (TM/LP) trip is designed to shut the reactor down should the reactor conditions (ASI, inlet temperature, core power and pressure) approach the point where DNB might occur during either normal operation or an AOO. The present analysis uses the HTP critical heat flux correlation and the statistical setpoint methodology described in Reference 1 and is consistent with the NRC's Standard Review Plan in requiring DNB to be avoided with 95% probability at a 95% confidence level.

The uncertainties shown in Tables 15.6-2 and 15.6-4, and the transient biases in Table 15.6-3 are included in the verification of the TM/LP trip as described in Reference 1. Axial power profiles and scram curves for the current cycle were included in this analysis. An excess margin of protection is provided by the existing trip for the current cycle.

15.6.5.2 Limiting Conditions for Operation

DNB Monitoring

The TM/LP trip system does not monitor reactor coolant flow and does not consider changes in power peaking which do not significantly change ASI. Thus, the TM/LP trip generally does not provide DNB protection for the four-pump coastdown and CEA drop AOOs. The LCO presented here administratively protects the DNB SAFDL for these transients.

The method used to establish the DNB LCO involved simulations of the CEA drop and the loss-of-flow transients using the core thermal hydraulic code XCOBRA-IIIC(10), to determine the initial power, as a function of ASI, which provides protection from DNB with 95% probability. The uncertainties listed in Tables 15.6-5, 15.6-6 and 15.6-7 are applied using the methodology described in Reference 1 and neutronics data. The statistical analysis accounted for the effects of uncertainties associated with incore operating parameters, the HTP critical heat flux correlation, and power peaking. Axial power profiles and scram curves for the EPU were included in the analysis. This analysis verifies the adequacy of the DNB LCO tent.

Linear Heat Rate Monitoring

The plant Technical Specifications allow plant operations for limited periods of time with the incore detectors out of service. In this situation, the LPD LCO barn provides protection in steady-state operation against penetration of the LPD limit established by LOCA considerations. The statistical methodology for the LPD LCO is essentially the same as that for LPD LSSS except the uncertainties listed in Table 15.6-5 were used, as opposed to the values in Table 15.6-1.

The allowed power versus ASI was statistically analyzed to account for the appropriate uncertainties. This analysis demonstrates the adequacy of the LPD LCO tent. The LHR protected by the LPD LCO is depicted in Figure 15.6-7.

15.6.6 REFERENCES FOR SECTION 15.6

1. EMF-1961(P)(A) Revision 0, "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors," Siemens Power Corporation, July 2000.
2. XN-NF-75-21(P) (A), Revision 2, "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation," Exxon Nuclear Company, January 1986.
3. "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," EMF-92-153(P)(A), Revision 1, Siemens Power Corporation, January 2005.

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TABLE 15.6-1

Uncertainties Applied in LPD LSSS Calculations

Parameter	Value ^a
Engineering tolerance	$\pm 3\%$
Peaking uncertainty	7% ^b
Power measurement uncertainty	See Figure 15.6-8
Thermal power uncertainty	$\pm 9.308\%$ (of rated) ^c
LPD trip overshoot uncertainty	0.0%
LPD trip transient offset	$\pm 9.42\%$
ASI uncertainty	$\pm 6\%$

^a Unless otherwise noted the distributions are treated as normal, two-sided, and the uncertainty range represents 95% bound ($\pm 1.96 \sigma$)

^b Treated as normal, one-sided distribution at 95% probability (1.645σ)

^c The thermal power uncertainty includes the effects of calorimetric uncertainty, thermal power instrumentation channel uncertainty, and measurement uncertainties.

^d Not treated statistically; treated as a bias.

TABLE 15.6-2

Uncertainties Applied in TM/LP LSSS Calculations

Parameter	Value ^(a)
Engineering tolerance	± 3%
Peaking (F_r^T)	6.0% ^(b)
Power measurement uncertainty	See Figure 15.6-8
PZR Pressure uncertainty	± 40 psi
TM/LP trip uncertainty	± 155 psi
Trip biases	Table 15.6-4
Inlet coolant temperature	± 3.0°F
HTP correlation	See Reference 3
Flow measurement uncertainty	± 15,000 gpm
ASI uncertainty	± 6%

^a Unless otherwise noted the distributions are treated as normal, two-sided, and the uncertainty range represents 95% bound ($\pm 1.96\sigma$).

^b Treated as normal, one-sided distribution at 95/95 probability (1.645σ).

TABLE 15.6-3

Transient Biases Applied in the TM/LP LSSS Calculation

Parameter	Stuck-Open PORV	UCRW at Power	Excess Load
Power (% of Rated)	0.19	-0.16	22.22
Pressure (psi)	4.70	25.20	-3.00
Cold Leg Temperature (°F)	0.06	-1.65	3.40
Hot Leg Temperature (°F)	1.13	3.92	0.43

(These biases account for differences between the measured parameter and actual core conditions during each transient.)

TABLE 15.6-4

Additional Trip Functions
(Only Those Used in Setpoint Verification)

Parameter (Database Keys)	Value Setpoint	Uncertainty ^(a)
Low reactor coolant flow	95.0%	4%
High pressurizer pressure	2400 psia	40 psi ^(b)
VHPT	9.61% of rated (offset) 107% of rated (ceiling)	See Footnote ^(c)

^a The distributions are treated as normal, and the uncertainty range represents a two-sided 95% bound ($\pm 1.96\sigma$).

^b The high pressurizer pressure trip uncertainty applied for Setpoint verification is conservative bounding of the required 35 psi uncertainty.

^c A combination of various uncertainties are applied to the VHPT setpoint for the LPD LSSS and LPD LCO setpoint verifications.

TABLE 15.6-5

General Uncertainties Applied in the LCO Calculations

Parameter	Uncertainty Value ^(a)
Engineering tolerance	$\pm 3\%$
Peaking uncertainty (F_r^T)	6.0% ^(b)
Flow measurement	15,000 gpm
Pressure measurement	± 40 psi
T_{inlet}	± 3.0 °F
Power measurement	See Figure 15.6-8
ASI	+ 6%
HTP correlation	See Reference 3

^a Unless otherwise noted, the distributions are treated as normal, two-sided, and the uncertainty range represents 95% bound ($\pm 1.96 \sigma$).

^b Treated as normal, one-sided distribution at 95/95 probability (1.645σ).

TABLE 15.6-6

Additional Uncertainties Applied in DNB LCO CEAD Calculations

Parameter	Uncertainty Value ^(a)
CEAD T _{inlet}	± 0.0°F
CEAD Pressure	± 0.0 psid

^a Unless otherwise noted the distributions are treated as normal, two-sided, and the uncertainty range represents 95% bound ($\pm 1.96 \sigma$).

TABLE 15.6-7

Additional Uncertainties Applied in DNB LCO LOCF Calculation

Parameter	Uncertainty Value ^(a)
Total peaking	7% ^(b)
CEA holding coil delay	0.0 s ^(c)
LOCF trip	0.040000 of rated ^(b)
Pump coastdown coefficient	0.007469 1/s ^(b)
Scram-speed scale factor	0.0 ^(c)
Scram worth	±0.0% ($\Delta k/k$) ^(d)

^a Unless otherwise noted, the distributions are treated as normal, two-sided, and the uncertainty range represents 95% bound ($\pm 1.96 \sigma$).

^b Treated as normal, one-sided distribution at 95/95 probability (1.645 σ).

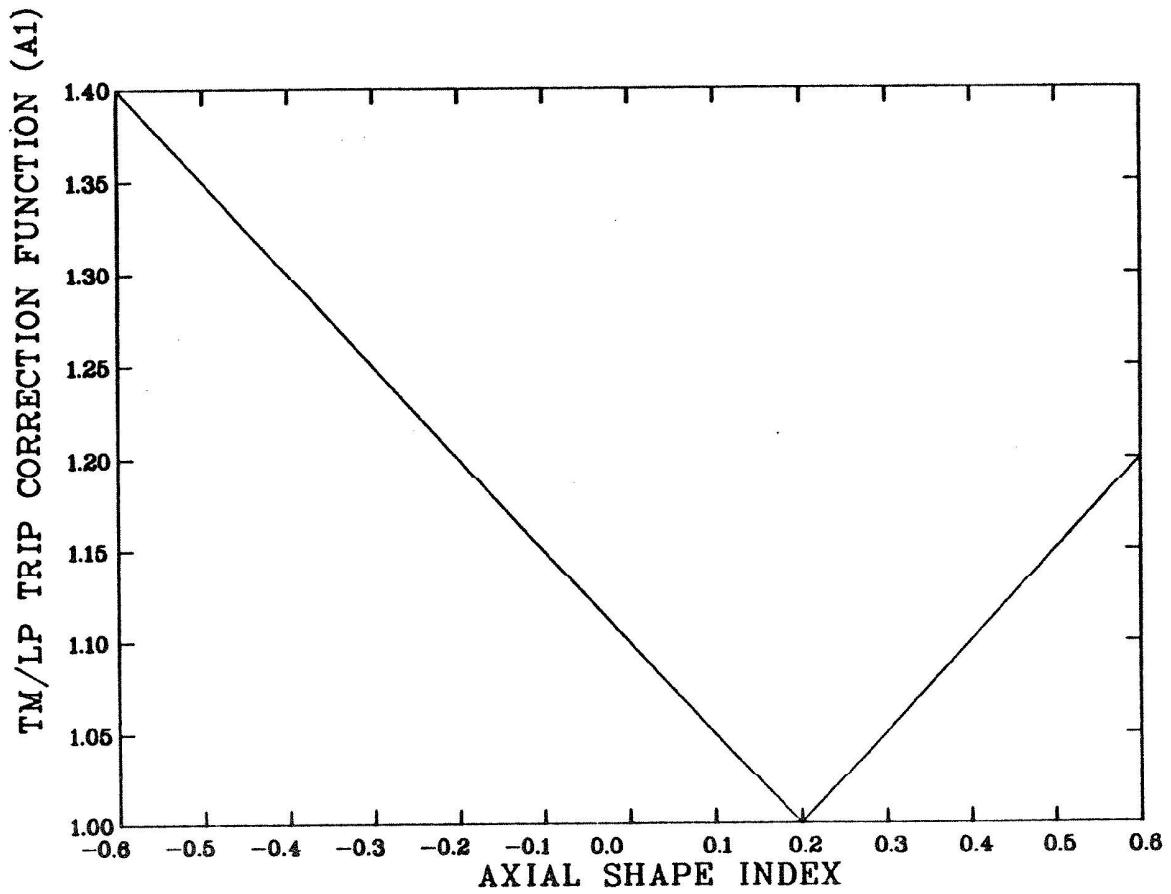
^c A deterministic approach is conservatively being used for the EPU due to the age of the supporting surveillance test data. The uncertainty in the rod drop and clutch coil delays are being treated deterministically which drives this uncertainty to zero.

^d Bounding value includes 10% uncertainty allowance.

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Figure 15.6-1

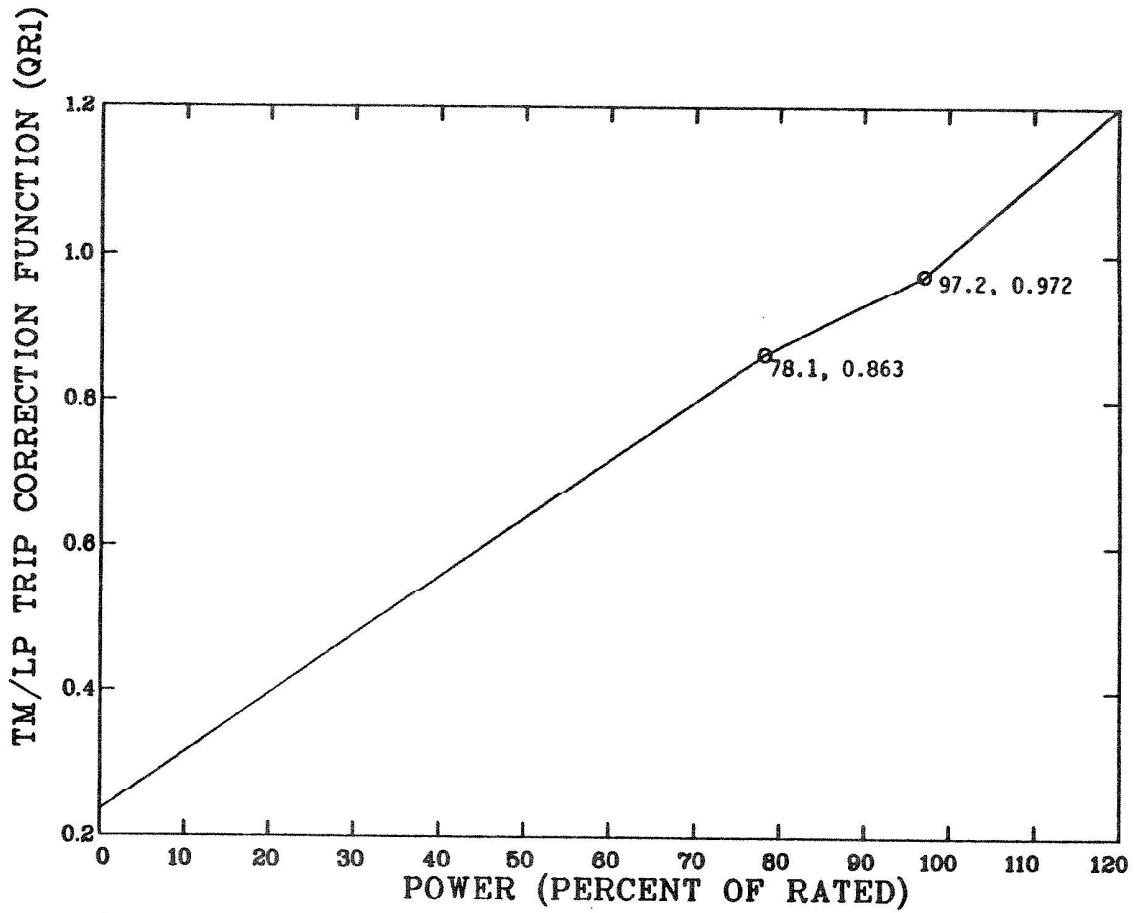


AMENDMENT NO. 7 (7/88)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

ST. LUCIE UNIT 1,
TM/LP TRIP FUNCTION A1

FIGURE 15.6-2



AMENDMENT NO. 7 (7/88)

FLORIDA POWER & LIGHT COMPANY
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ST. LUCIE UNIT 1,
TM/LP TRIP FUNCTION QR1

FIGURE 15.6-3

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Figure 15.6-4

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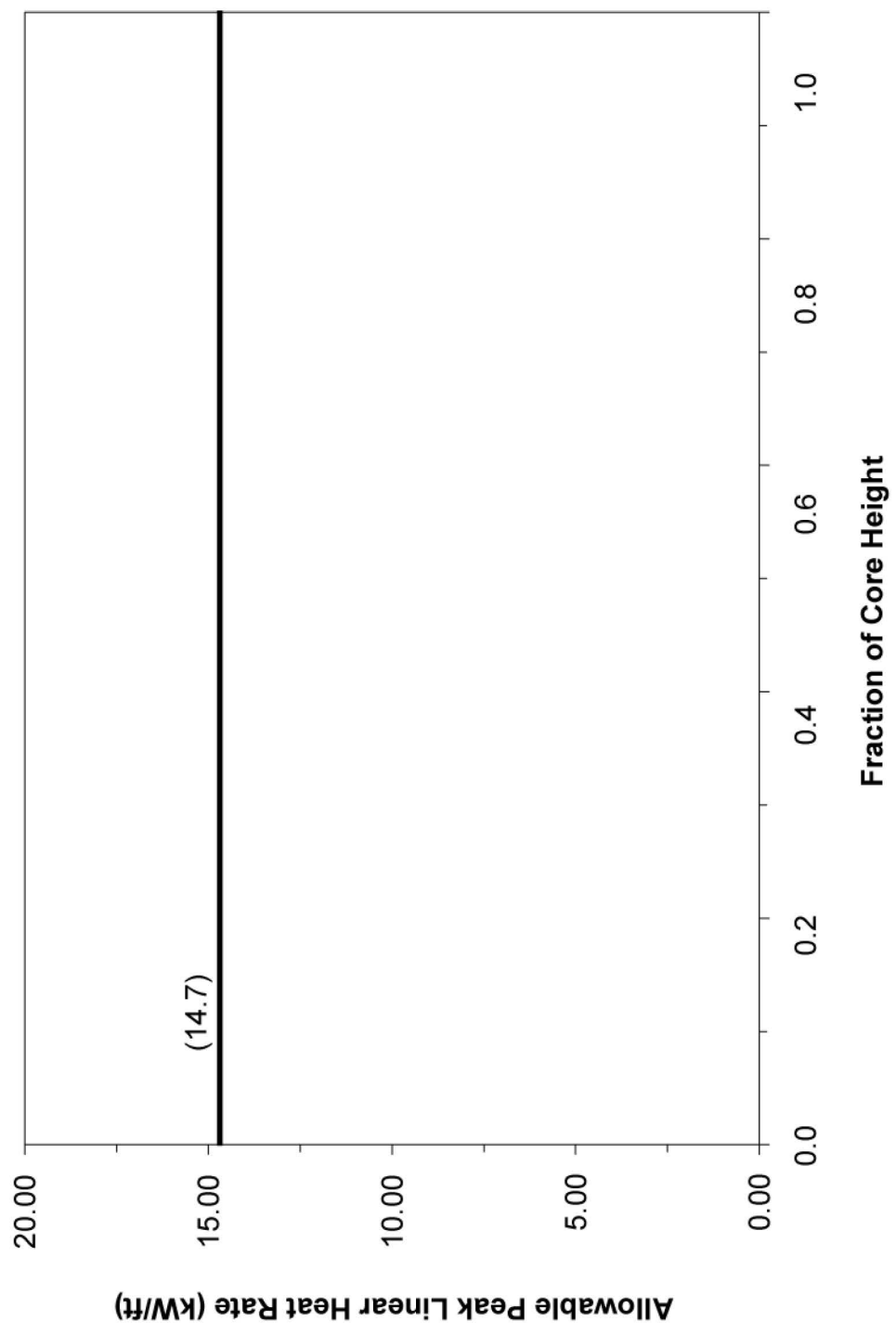
Figure 15.6-5

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Figure 15.6-6

Linear Heat Rate LCO Used in LPD LCO Verification

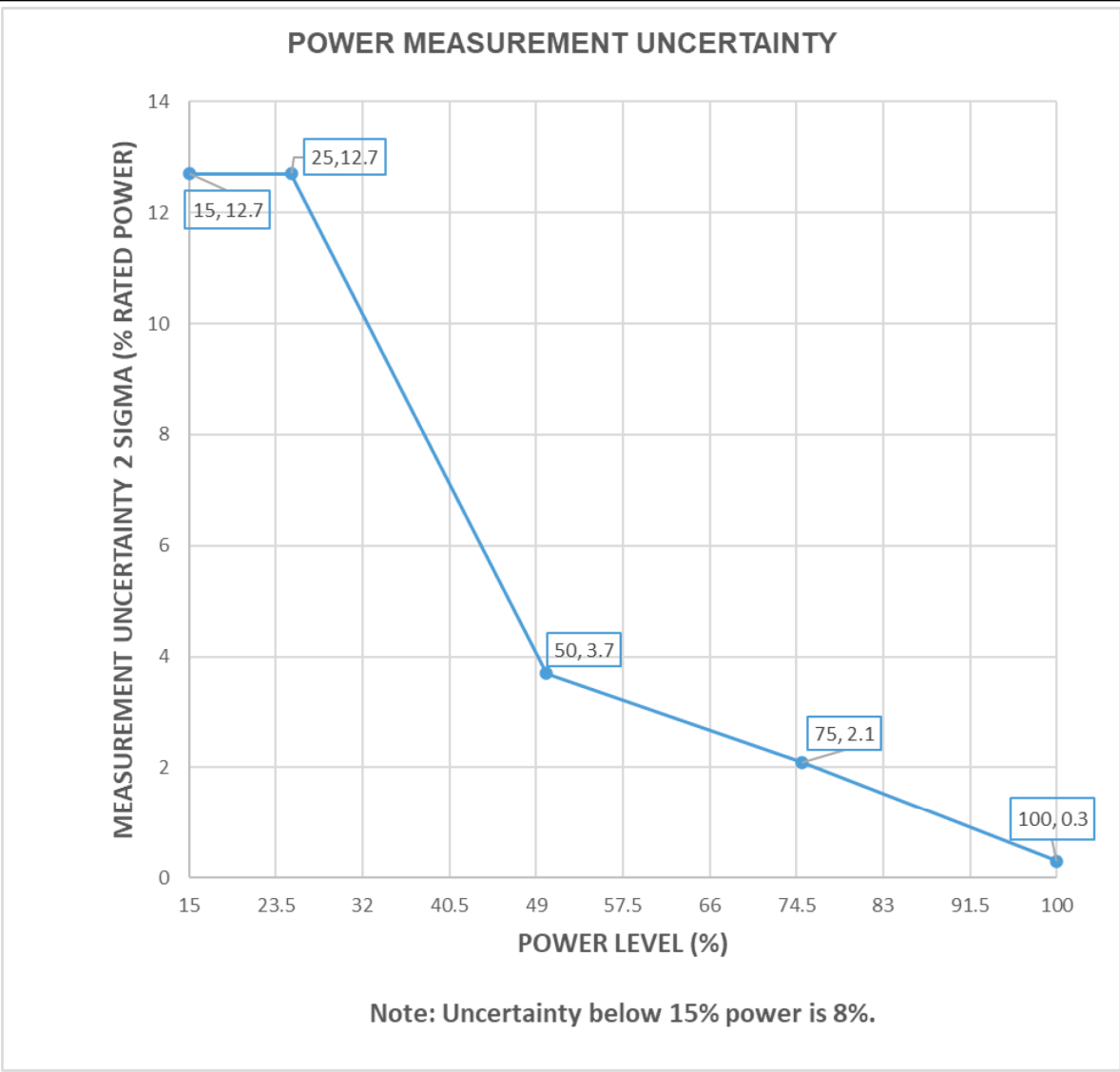


Amendment No. 26 (11/13)

**Florida Power & Light Company
St. Lucie Plant Unit 1**

**LINEAR HEAT RATE LCO USED IN LPD
LCO VERIFICATION**

FIGURE 15.6-7



Amendment No. 33 (10/24)

**Florida Power & Light Company
St. Lucie Plant Unit 1**

**POWER MEASUREMENT UNCERTAINTY
VS. POWER**

FIGURE 15.6-8